August 21, 2003

MEMORANDUM TO: Chairman Diaz
Commissioner McGaffigan
Commissioner Merrifield

FROM: William D. Travers **/RA/William Travers**
Executive Director for Operations

SUBJECT: UPDATED PROGRAM PLAN FOR HIGH-BURNUP LIGHT-WATER REACTOR FUEL

Fuel rod cladding is the first barrier for retention of fission products, and the structural integrity of the cladding ensures a coolable core geometry. In the early 1990s, new data from foreign research programs showed degraded cladding behavior for high-burnup fuel under certain postulated accident conditions. It thus became clear that extrapolation from a low-burnup database needed to be reassessed for regulatory purposes. In a July 6, 1998, memorandum, the staff informed the Commission of an agency program plan to address issues related to the use of high-burnup fuel and a strategy for assessing future requests for burnup extensions beyond the current NRC limit of 62 GWd/t burnup (average for the peak rod). Attached is an update of the plan.

The plan discusses fuel-related regulatory issues and documents that are undergoing regulatory reviews. The plan also addresses fuel damage limits that are used as regulatory criteria for maintaining a coolable core geometry. These criteria are used broadly to ensure that design-basis overpower and undercooling events do not progress into core melt scenarios. The criteria are also used in probabilistic risk assessments to determine whether certain sequences result in core damage. These criteria were all developed about 30 years ago and have not been adjusted for high-burnup fuel or for new cladding alloys that have been introduced to achieve high burnups.

One criterion that needs to be modified is the 280-cal/g fuel enthalpy limit that is used for reactivity insertion accidents (e.g., Regulatory Guide 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors”). Based on preliminary information from our research program, a substantially reduced interim value of around 100 cal/g is being used in some technical assessments. Research on this subject is advancing quickly and we expect to resolve the issue by December of this year.

CONTACT: Ralph O. Meyer, RES
415-6789

Another criterion that needs attention is the 17-percent oxidation limit in 10 CFR 50.46 ("Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors") for analysis of loss-of-coolant accidents (LOCAs). Niobium, which is used in advanced cladding alloys, can behave very differently than the tin in current alloys. Some niobium-bearing cladding alloys have been found to require an oxidation limit as low as 6 percent to ensure coolable geometry. Fortunately, the specific products currently used in the U.S. are compatible with the 17-percent limit, but this situation is not well-controlled at this time. Current research is expected to provide a basis for a performance-based modification to the oxidation criterion in 10 CFR 50.46 by mid-2004. Confirmation for high-burnup advanced alloys will be provided at a later date, subject to the availability of fuel rods and the continued cooperation of the industry.

Yet a third criterion that needs to be reviewed is the cladding temperature limit used for dry cask storage and transportation. Particularly during a vacuum drying procedure, high temperatures can lead to embrittlement of the cladding and the cladding may not meet confinement requirements (e.g., 10 CFR 72.122(h), "Confinement barriers and systems"). Based on preliminary information, this temperature limit has been reduced from 570 °C to 400 °C in interim licensing guidance. The test results for the final evaluation of Zircaloy cladding will be available later in 2003, and confirmation for high-burnup advanced alloys will be provided at a later date, subject to the availability of fuel rods and the continued cooperation of the industry.

NRC high-burnup fuel activities address reactor events, spent fuel issues, and analytical methods. NRC's research is done at three national laboratories and at foreign facilities in accordance with five international agreements. The research program is currently funded at approximately 2.8 FTE and $4.7M. In the next few years we expect to develop the technical basis to confirm previous regulatory decisions and to review new industry initiatives involving advanced cladding. The largest single expenditure in the research program is for testing high-burnup fuel rods from commercial plants in the hot cells at Argonne National Laboratory. This work is being done in cooperation with the Electric Power Research Institute, Framatome ANP, Westinghouse Electric Company, and the Department of Energy. About two-thirds of the expenditure at Argonne is in the reactor arena and about one-third is in the waste arena. The NRC also stretches its resources by participating in international projects worth more than $30M per year. We believe that the level of expenditure is justified because the data will enable NRC to ensure the safe utilization of high-burnup fuel.

The major results of the fuel research to date are as follows:

- Preliminary test results used to provide interim criterion for reactivity accidents
- Completion of an adequate set of reactivity insertion tests for issue resolution
- Basis for NRC's position on corrosion and 10 CFR 50.46 oxidation limit
- Peculiar behavior of niobium alloys identified and pointed out to NRR
- Basis for interim staff guidance (ISG-11, Rev. 2) for spent fuel cask evaluation
- Basis for interim staff guidance (ISG-8, Rev. 2) for spent fuel burnup credit
- Preliminary results used in draft resolution of GSI-185 (a boron dilution event)

The expected results of the research are described in more detail in the attachment:

- Resolution of reactivity accident issue for all cladding alloys, 12/03
- Basis for performance-based criteria for 10 CFR 50.46 on LOCA, 6/04
- Revision of Regulatory Guide 1.77 on reactivity accidents, 12/05
- Confirmation of revised 10 CFR 50.46 for advanced alloys, 12/06 and 12/08
- Resolution of boiling water reactor power oscillation issue, 2005-06
- Technical basis (cladding behavior and isotopic content) for spent fuel reviews, 12/03 (older alloys), 12/06 and 12/08 (advanced alloys)

The research activities and methods for resolving the technical issues mentioned above were reviewed by the Advisory Committee on Reactor Safeguards (ACRS) in October 2002. The ACRS sent a favorable letter on its review to the Chairman on October 17, 2002. Additional details on our high-burnup fuel activities are given in the attachment.

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UPDATED PROGRAM PLAN
FOR HIGH-BURNUP LIGHT-WATER REACTOR (LWR) FUEL

1 Introduction

There are currently 103 operating commercial nuclear power plants in the United States. All of the plants use uranium oxide fuel clad in cylindrical tubes of zirconium-based alloys. This fuel, with the fission products it contains, is a central focus of the NRC's reactor safety activities. Transportation and storage of spent fuel in dry casks are also regulated by the NRC. The ultimate disposal of this fuel is also a significant undertaking of the United States, through the Department of Energy (DOE), with license approval required from the NRC.

In the 1970s, when most of the licensing criteria and related analytical methods (computer codes) were being established, plant burnups were expected to be no more than about 40 GWd/t (average for the peak rod). Data out to that burnup had been included in databases for criteria, codes, and regulatory decisions, and it was believed that some extrapolation in burnup could be made. In the early 1980s, the industry requested increases in burnup for various levels up to 60 GWd/t based on compliance with existing fuel damage limits, and the NRC granted those requests. By the mid 1980s, however, unique changes in pellet microstructure (the rim effect) had been observed at higher burnups, along with increases in the rate of cladding corrosion (breakaway oxidation). It thus became clear that something new was happening at high burnups (i.e., above 40 GWd/t) and that extrapolation from the low-burnup database could not continue indefinitely.

In a July 6, 1998, memorandum, the staff informed the Commissioners of an agency program plan to address issues related to the use of high-burnup fuel and a strategy for assessing future requests for burnup extensions beyond the current limit of 62 GWd/t burnup (average for the peak rod), referred to as extended burnup. The plan summarized a broad range of fuel issues that were being actively investigated at NRC at that time. The plan also laid out specific industry responsibilities for proposals to extend burnups beyond the current limit.

In the original program plan, a list of high-burnup issues was identified based on observed operational problems, experimental results from test programs, and an understanding of basic phenomena. That list is shown in Table 1. The discussion of each issue in the program plan included (a) a description of the issue (the origin of the concern raised by high-burnup operation); (b) a risk perspective on the issue; (c) a near-term assessment of why it was satisfactory to wait 3-5 years, in some cases, for research results to achieve final resolution; (d) a description of related NRC research; and (e) a description of what would constitute final resolution.

Table 1. High-Burnup Issues in the 1998 Program Plan

ISSUE	STATUS
Cladding Integrity and Fuel Design Limits	Resolved[a]
Control Rod Insertion Problems	Resolved[a]
Criteria and Analysis for Reactivity Accidents	Active
Criteria and Analysis for Loss-of-Coolant Accidents	Active
Criteria and Analysis for BWR Power Oscillations (ATWS)	Active
Fuel Rod & Neutronic Computer Codes for Analysis	Resolved[a]
Source Term and Core Melt Progression	Resolved[a]
Transportation and Dry Storage	Active
High Enrichments (>5%)	Deferred

[a]Initial objectives have been met, but follow-on activities may be ongoing (see text).

The first two issues, related to cladding integrity, fuel design limits, and control rod insertion, were described as being satisfactorily addressed by industry activities for current fuel designs and the current burnup limit of 62 GWd/t. The last issue, high enrichments, remains a possible future action with no near-term activity needed with respect to high-burnup fuel.

To help determine which of the remaining issues warranted greater efforts for resolution, risk concepts were employed in the original plan. Of course, consideration of compliance and defense-in-depth also affected that determination, and a balance was sought in pursuing each issue. This document updates the status of those issues, and it also describes other licensing and regulatory activities at NRC related to the current burnup limit and to future burnup extensions. Fuel-related activities associated with severe accidents (except source terms), the use of mixed-oxide (MOX) fuel, and advanced reactors other than light-water reactors are described in other documents. Those activities, such as the recent work on low-temperature air oxidation of fuel rod cladding, are well coordinated with the work described here and do not duplicate any of these efforts.

2 Licensing and Research Strategy

In general, NRC establishes criteria and licensees show compliance with those criteria. For high-burnup fuel, the NRC's criteria and the licensee's analyses may be affected by the level of burnup. To assess possible effects, the staff is following the course set out in the July 6, 1998, memorandum. (1) Information has been provided (and is repeated here) to show why it is acceptable to wait for well documented confirmatory assessments of existing approvals to operate to 62 GWd/t burnup. (2) For burnups up to the current limit of 62 GWd/t, the staff is

performing research and assessing the adequacy of NRC criteria and existing licensing analyses in a confirmatory activity. (3) For burnups above the current limit, the staff will determine what information must be provided by the applicant as part of their license application, and what additional NRC research is needed to support the licensing offices in their review of proposed revisions to NRC criteria and licensee-submitted compliance analyses.

Transportation and storage of spent fuel of a given burnup takes place significantly later than the operation of reactors with fuel of that burnup. Therefore, the high-burnup issues related to fuel and cladding behavior in reactors were discovered before licensing decisions had been made to approve casks for high-burnup fuel. This information was thus routinely incorporated into reviews for cask license extensions, and confirmation of previous licensing decisions has not been needed.

In accordance with the NRC's Strategic Plan, the staff has encouraged the industry to develop codes, standards, guides—and, by inference, fuel damage criteria—that could be endorsed by the NRC. In the past, fuel damage criteria were developed by the NRC. It is expected, however, that any research to support such regulatory criteria will be nonproprietary, to ensure that the resulting criteria can be fully justified in a public forum. The NRC staff expects to have full access to, and may actively participate in, those research programs. Fuel behavior must be addressed for reactors (normal operation, transients, and postulated accidents) and dry casks. At a minimum, the high-burnup issues identified above must be covered for extended-burnup applications. At the present time, the NRC is engaged in cooperation with the industry in the data phase of several test programs, and is seeking more such opportunities to obtain important data for its own independent assessment.

To develop the database necessary to justify extended burnup, suitable fuel rod specimens must be available for testing under reactor (transient and accident) and dry cask conditions. For this purpose, the NRC has encouraged the irradiation of lead test assemblies (LTAs) with typical burnup histories up to the proposed licensing limit and positioned in near-limiting core locations. The staff has approved new guidance to simplify the regulatory approval process for irradiation of extended-burnup LTAs. A program for monitoring fuel performance in reactors is also important in any industry proposal.

3 Staff Activities To Address the Utilization of High-Burnup Fuel

In addition to research that was initiated a few years ago on the issues in the original program plan, several related industry submittals have been made and more are expected. These include requests for extended burnup of LTAs in reactors beyond the present 62 GWd/t limit and increases in burnup in casks for transportation and dry storage. These requests generally involve the use of advanced cladding alloys — particularly niobium-bearing alloys — that were designed for better resistance to corrosion to achieve very high burnups. Hence, a shift in NRC programs is underway from work on Zircaloy cladding alloys to work on the ZIRLO and M5 cladding currently in use for PWRs. In the following sections, major staff activities related to current burnups and extended burnups are described. These activities address the issues identified in the original program plan and are described in the order of that plan.

3.1 Fuel Designs For High-Burnup and Extended-Burnup Cores

NRC performs reviews of licensing actions that are requested by licensees with respect to individual power plants, as well as reviews of generic topical reports that are prepared by reactor vendors and industry groups to address issues in a generic manner. The following major licensing actions and topical report reviews are related to the approval of new fuel designs for existing high-burnup cores, and for extended burnup. These reviews have recently been completed, are in progress, or are expected to be submitted for review in the near future. Other submittals, which have not been announced, are also expected. The activities described here address the original issues of cladding integrity, fuel design limits, and control rod insertion.

Table 2. LWR Fuel Licensing Activities

Applicant	**Description**	**Completion**
Westinghouse	Control Rod Ejection Method	5/30/2003 C
Westinghouse	Fuel Rod Design Methods	9/30/2003
GNF	High Exposure Fuel Rod Thermal-Mechanical Model	3/01/2004
GNF	Frequency of Densification Sampling	3/30/2003 C
Framatome	Mark BW Burnup Extension	1/31/2002 C
Framatome	Fuel Mechanical Design for Advanced Mark BW	4/30/2003 C
Framatome	M5 Incorporation in ANP Approved Methods	6/30/2003
Westinghouse	Byron LTA Exemption Request	9/30/2003
GNF	Revised Creep Collapse Criterion	3/30/2003 C
Westinghouse	Optimized (Low-Tin) ZIRLO Cladding	5/31/2004

3.2 Criteria and Analysis For Reactivity Accidents

The specific accidents of interest are the rod ejection accidents in a PWR and the rod drop accident in a BWR. For these postulated accidents, the NRC has 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 for cladding failure (departure from nucleate boiling, minimum critical power ratio, and 170 cal/g peak fuel enthalpy) for the purpose of dose calculations. These requirements and criteria are described in General Design Criterion 28, Regulatory Guide 1.77, and Standard Review Plan 4.2. 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] Figure 1 shows such a cladding failure in a recent test in the Nuclear Safety Research Reactor at the Japan Atomic Energy Research Institute.[2] Thus,

cladding failure -- sometimes with fuel dispersal -- can occur at significantly lower energies than previously thought, and the current criteria need to be revised.

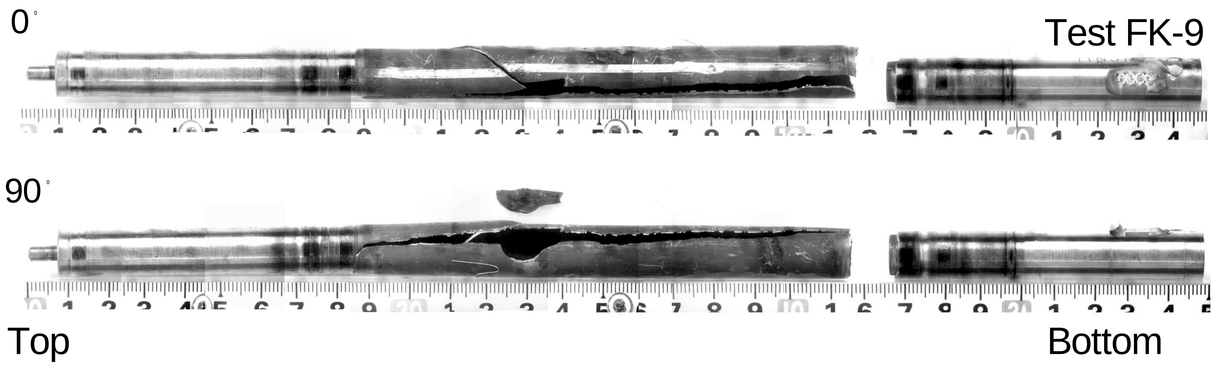

Figure 1. Partially brittle cladding failure in a reactivity transient test with a high-burnup Zircaloy-clad BWR fuel rod

From the general discussion of risk in the original program plan, it was seen that the frequency of occurrence of a BWR rod drop accident is below the range of interest for consideration as a generic issue, whereas the frequency for a PWR rod ejection accident is just within that range. Therefore, it was concluded that these events could be analyzed in a best-estimate manner rather than adding substantial conservative margin, but accurate criteria and neutron kinetics methods should be used.

(a) Confirmatory Assessment For Burnups up to 62 GWd/t

Prior to preparing the original program plan, the staff reached a preliminary conclusion that it was likely that peak fuel enthalpies in LWRs would remain below the lower enthalpy values associated with cladding failure in then-recent tests. More recent data and analyses continue to support that preliminary conclusion.

There has been no test program of this kind in the U.S. for over 15 years, so the NRC entered into formal agreements with France (the Cabri test reactor), Japan (the NSRR test reactor), and Russia (the IGR and BIGR test reactors) to obtain data from current programs. Those programs have now produced a substantial amount of new data, as can be seen in Fig. 2.[3 4] However, the data as plotted exhibit a lot of variation such that additional interpretation of the data is required. This data scatter is an indication that other variables besides oxidation are affecting fuel enthalpy at failure. Those variables include pulse width, test temperature, burnup, and cladding mechanical properties.

During 1999–2001, the staff developed phenomenon identification and ranking tables (PIRTs) for fuel behavior under reactivity accident conditions with the help of a large panel of international experts.[5 6] Concepts for dealing with variables that affect test results came out of that exercise. At this time the staff is pursuing three methods of interpreting the test data for LWR conditions. The first involves calculating strain (or, equivalently, strain energy density) under LWR conditions with NRC's FRAPTRAN fuel rod code after validating the code with the

test data. Those calculations would then be compared with measured total elongation (or critical strain energy density) from mechanical properties measurements to determine when failure would occur. The second method involves correcting (or scaling) the data points in Figure 2 for LWR pulse widths and temperatures using FRAPTRAN and mechanical properties data. Curve-fitting techniques would then be used to find the best curve that separates failures from nonfailures. The third method involves using a multiparameter empirical correlation such as recently developed by Vitanza to predict failure for LWR conditions.[7] We expect that one or more of these methods will be successful and provide a good estimate of the fuel enthalpy at which cladding failure could be expected in an LWR. This result will then be compared with plant calculations to determine acceptability.

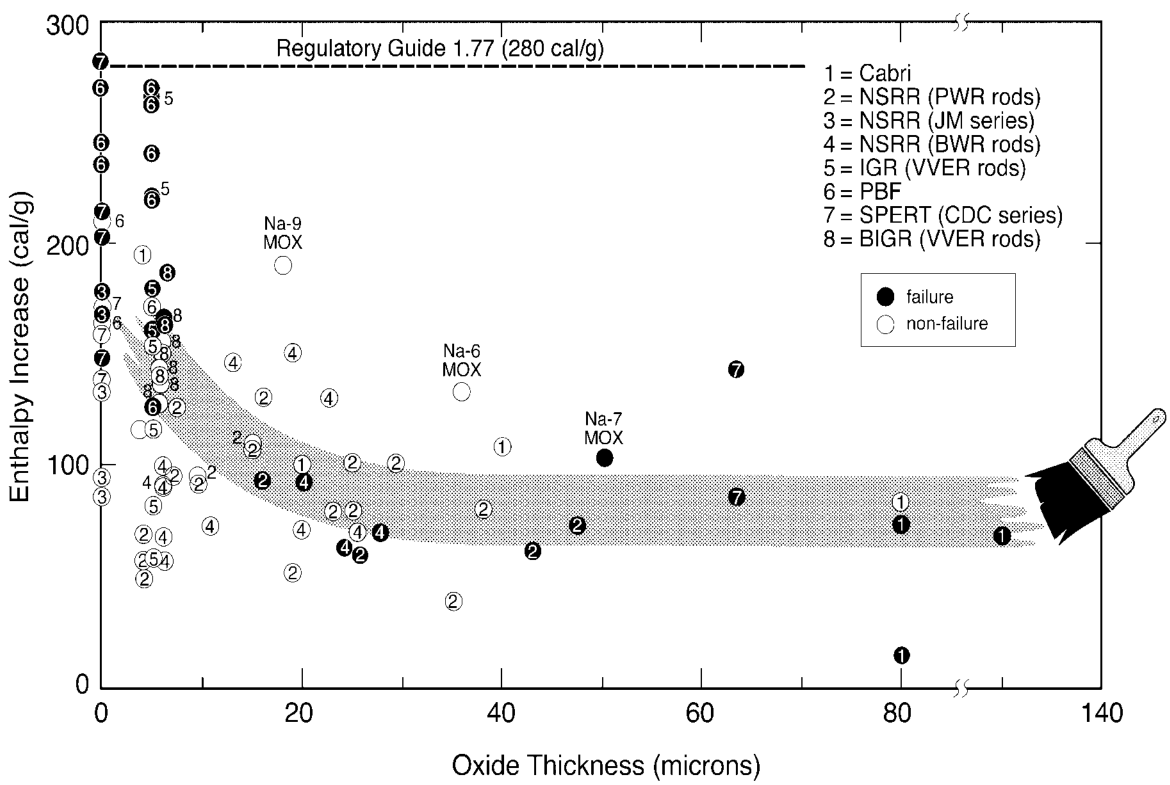

Figure 2. Database for reactivity-initiated accidents showing cladding failures well below the current limit

Generic plant calculations are being done to obtain expected energy deposition for postulated reactivity accidents. NRC's PARCS three-dimensional neutron kinetics code is being used for these calculations for a range of typical core designs. Peak fuel enthalpies from the calculations are being obtained to compare with the cladding failure threshold. Our preliminary calculations, and the calculations of 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.[8] To apply these results to a broad class of operating reactors, control rod worths will be examined. Control rod worths needed to reach the cladding failure threshold will be determined from the generic plant calculations, and the rod worths will be compared with rod worths in operating reactors. Some screening of control rod worths in current commercial operating cycles will be

performed to complete the assessment . Based on preliminary results, it is expected that the rod worths will be less than the rod worth needed to reach the fuel enthalpy for cladding failure. Therefore, we continue to believe that this accident will result in little or no fuel damage and that we can wait a little longer for a well-documented confirmatory assessment for the current burnup limit.

The above analytical and experimental work is underway at this time. Utilizing results from that work, the current schedule for completing a confirmatory assessment is December 2003 for all currently approved cladding types. At that time, we plan to issue a Research Information Letter with a well-documented safety analysis to confirm that current operating reactors meet the intent of General Design Criterion 28 for currently approved cladding types and burnups up to 62 GWd/t. This will resolve the issue identified in the original program plan. The data and analyses that are being developed in this research effort can also be used in assessing the risk associated with other non-design-basis reactivity events.

(b) Review of Industry Assessment for Burnup Extension From 62 to 75 GWd/t

In accordance with the licensing and research strategy described above and in the original program plan, the Electric Power Research Institute (EPRI) has submitted proposed revisions to the regulatory criteria for reactivity accidents.[9] If accepted by NRC, these proposed criteria will be applied to all LWR fuel for burnups up to 75 GWd/t, which encompasses both the current- and the extended-burnup ranges. The report is under review at this time. The schedule for completion of this review is June 2004, which is about 6 months after the completion of NRC's confirmatory work.

Following the completion of this review, revisions will be made to Regulatory Guide 1.77 and Standard Review Plan 4.2 to bring these guidance documents into conformance with the criteria that have been accepted. The estimate for this work is 18 months, including a period for public comment and ACRS review, with completion in December 2005. After completing these activities, the staff will continue to participate in long-term international research programs for further confirmation.

3.3 Criteria and Analysis for Loss-of-Coolant Accidents

For these postulated accidents (any break size that results in core uncovery), the NRC uses cladding embrittlement criteria in 10 CFR 50.46 (2200 °F peak cladding temperature, 17 percent maximum cladding oxidation) to ensure that coolable geometry is not lost. Related evaluation models, such as described in 10 CFR Part 50 Appendix K, are 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 and can experience some loss of ductility (embrittlement) from related hydrogen absorption. In a few cases, measured oxidation levels on Zircaloy cladding have been an appreciable part of the 17 percent limit (100 microns of oxidation is approximately 10 percent equivalent cladding oxidation). This pre-transient oxygen (and related hydrogen) from the burnup process will contribute to

embrittlement during the transient, although it is not clear that high-burnup fuel will become limiting.

Another source of hydrogen, which was not understood when the criteria in 10 CFR 50.46 were established in 1973, is the disassociation of water that takes place during oxidation on the inside of the cladding in the vicinity of a cladding rupture. Figure 3 shows a brittle crack that occurred in this high hydrogen region of an unirradiated test specimen.

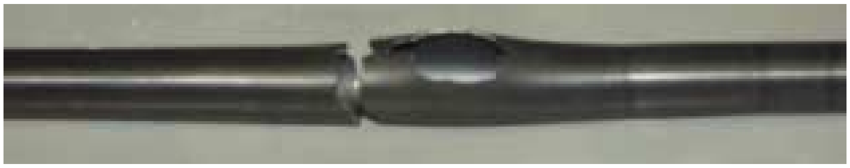

Figure 3. Brittle cladding failure due to hydrogen embrittlement in an unirradiated Zircaloy-clad specimen after a simulated LOCA transient

The core damage frequency attributed to large-break LOCA is quite small. This is based on the reliability of equipment modeled in the PRA and is predicated on the assumption that the fuel damage criteria are correct. The plan is intended to confirm the applicability of these criteria for high-burnup fuel. The criteria are also important for small-break LOCA, where cladding oxidation may be limiting.

Since the original high-burnup program plan was issued, the staff issued an information notice (IN 98-29) on the importance of pre-accident oxidation. Further, in late 1999, the staff clarified its position that total oxidation thickness should include the pre-accident oxidation for the purpose of comparison with the 17 percent limit in 10 CFR 50.46.[10] This clarified definition of total oxidation is now being used by all licensees, and the staff believes that the inclusion of pre-accident oxidation conservatively accommodates burnup effects on the embrittlement criteria in 10 CFR 50.46. Burnup may also affect the ballooning process and the oxidation kinetics, but early test results from NRC's research do not indicate large effects.

(a) Confirmatory Assessment for Burnups up to 62 GWd/t

The effects of high burnup on the embrittlement criteria and the fuel-related analytical models are being addressed in a program at Argonne National Laboratory. The program is sponsored by NRC with cooperation from EPRI and the Department of Energy. Framatome and Westinghouse also participate in a portion of this program dealing with unirradiated advanced cladding alloys. NRC participates in related (and coordinated) programs in Norway (Halden) and Russia (Kurchatov), and the staff has access to other related international work. These cooperative programs help inform the NRC research with regard to industry issues and leverage NRC resources.

The NRC staff also use computer codes to perform LOCA calculations. The FRAPCON-3 and FRAPTRAN codes are discussed in a later section (Section 3.5). Plant systems codes, which describe thermal, hydraulic, and neutronic behavior of the reactor, are not in the scope of this program plan; nevertheless, fuel-related models developed in this program will be fed into the plant systems codes as appropriate.

During 2000–2001, the staff developed phenomenon identification and ranking tables (PIRTs) for fuel behavior under LOCA conditions with the help of a large panel of international experts.[11] As with the reactivity accidents, the staff is utilizing the PIRT results to help structure the research at Argonne. A detailed outline of the work to address LOCA-related issues was presented recently by the staff.[12] This work involves the use of ring-compression tests for the embrittlement criteria and direct measurement of oxidation rates and ballooning behavior in high-burnup fuel rod specimens.

Resolution of the LOCA issue will be accomplished in stages. Confirmatory assessment for fuel clad with Zircaloy-2 and Zircaloy-4 is scheduled for completion in December 2004. This assessment will be based primarily on work at Argonne on high-burnup BWR and PWR specimens of Zircaloy-clad fuel that are on hand. Figure 4 shows the first-ever test to be performed under LOCA conditions on a high-burnup fuel rod.

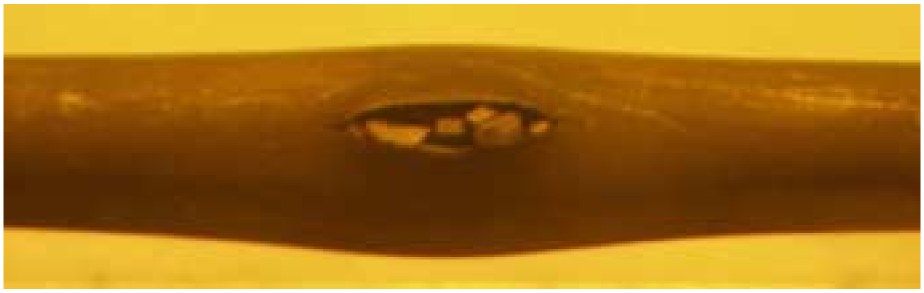

Figure 4. First-of-a-kind burst test with a high-burnup Zircaloy-clad BWR fuel rod specimen

ZIRLO and M5 cladding alloys contain niobium to reduce cladding corrosion during normal operation. However, the presence of niobium can significantly alter the behavior of the cladding under accident conditions. In these alloys, niobium does not go into complete solution in zirconium (as tin does in Zircaloy), and this leads to a much more complex microstructure containing precipitates. This microstructure can affect the morphology of the oxide that forms at high temperatures, and the oxide morphology in turn can affect the propensity for hydrogen absorption, which contributes to embrittlement. Figure 5 shows the non-protective oxide that can form on some niobium-bearing alloys, in contrast to the protective oxide that forms on other similar niobium-bearing alloys, as shown in Figure 6.

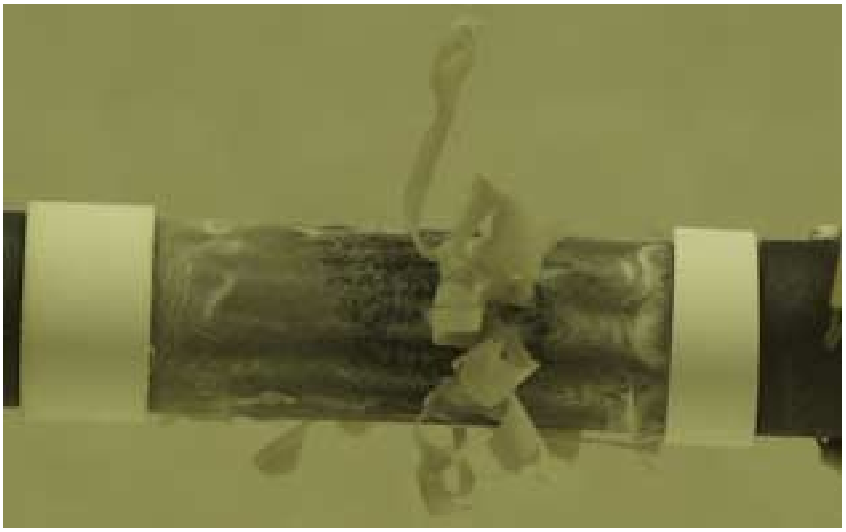

Figure 5. Non-protective oxide appears on E110 cladding (Zr-1%Nb) before ~10% oxidation in steam at 1000 °C

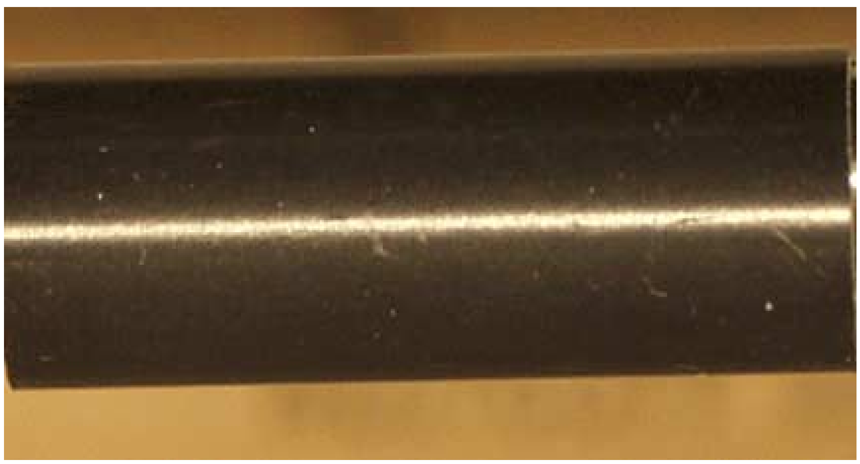

Figure 6. Protective oxide remains on M5 cladding (Zr-1%Nb) after ~17% oxidation in steam at 1000 °C

Although tests by Westinghouse and Framatome show that their unirradiated ZIRLO and M5 alloys embrittle about the same as Zircaloy under LOCA conditions, other similar niobium-bearing alloys do not (Russian E110 and E635).[13] The reasons for this different behavior are not yet understood, but are likely related to the complex nature of zirconium-niobium alloys. Thus, tests of these cladding types are being planned for a range of burnups to see if irradiation, corrosion, and hydrogen uptake during normal operation affect their behavior under LOCA conditions. If such fuel specimens can be acquired by December 2004, it will take

approximately 2 more years (until December 2006) to complete an assessment of ZIRLO-clad fuel. It will take 2 more years (until December 2008) to complete an assessment of M5-clad fuel, provided that such fuel specimens are acquired in a timely way. Continued industry cooperation is being sought to complete this work.

(b) Review of Industry Assessment for Extended Burnup Above 62 GWd/t

Available extended-burnup data are limited at this time for fuel behavior under LOCA conditions, and the industry has not made any submittals on this subject. However, in accordance with the licensing strategy, which was adopted in the original program plan, the industry is expected to address this issue. Therefore, submittals and a corresponding review effort are expected.

On March 31, 2003, a staff requirements memorandum (SRM) on SECY-02-0057 was issued that included a performance-based revision to the embrittlement criteria in 10 CFR 50.46. The staff plans to complete this revision to section 50.46 by December 2004 — a date that corresponds to the completion of the confirmatory assessment for fuel clad with Zircaloy-2 and Zircaloy-4. By that time, NRC's research should yield data on burnup effects during a LOCA for Zircaloy 2 and Zircaloy 4 cladding materials. During that same period, other research in the same program should yield data on alloy effects for unirradiated ZIRLO and M5 cladding. Based on those limited data on burnup and alloy effects, the staff plans to prepare performance-based embrittlement criteria that can be applied to all alloys at all burnups. The staff will also confirm the applicability of the current prescriptive criteria for Zircaloy, ZIRLO, and M5 (viz., 2200 °F peak temperature, 17 percent maximum oxidation) or develop new ones in light of the new research. Both the performance-based and the prescriptive criteria will need to be confirmed for high-burnup fuel with ZIRLO and M5 cladding, which might not behave like Zircaloy. That confirmation is contingent on receiving irradiated fuel rods from power plants and on continuing research cooperation from the industry.

3.4 Criteria and Analysis for BWR Power Oscillations (ATWS)

Section 50.62 of 10 CFR describes requirements to prevent or terminate power oscillations that may be associated with anticipated transients without scram (ATWSs) in BWRs. Because ATWS is an accident that is beyond the design basis, this regulation, unlike the LOCA rule, does not contain any criteria to ensure coolable fuel geometry. However, it is assumed for risk-assessment purposes that core melt will not occur if the equipment described in the rule functions as designed.

(a) Confirmatory Assessment for Burnups up to 62 GWd/t

In General Electric's original assessment of fuel integrity during an ATWS with oscillations, calculated fuel cladding temperatures compared favorably with the 2200 °F peak cladding temperature limit for LOCAs.[14] In a more recent assessment by General Electric, which was approved by NRC, cladding temperatures were found to exceed 2200 °F in some cases.[15] Nevertheless, it was concluded that core damage would not occur because the fuel enthalpy remained below the 280 cal/g fuel enthalpy limit for reactivity accidents. The staff's review in the original high-burnup program plan was based on a critique of the 280 cal/g limit. Subsequently, however, the staff discussed this accident sequence with a group of fuel experts who were

constructing phenomenon identification and ranking tables (PIRTs) for this accident sequence.[16] Based on those discussions, the staff now believes that high-temperature LOCA-like cladding embrittlement could be the most likely cause of loss of coolable geometry for this type of accident. From General Electric's more recent calculations, some of these ATWS sequences are seen to exceed the LOCA cladding temperature limit, and coolable geometry would then not be assured. Consequently, some event sequences for which oscillations were terminated according to 10 CFR 50.62, might loose coolable geometry before the oscillations were terminated.

The staff maintains a small research effort to try to understand these events more thoroughly. The staff has access to the results of recent repeated-pulse tests in the Japan Atomic Energy Research Institute's NSRR test reactor and will receive the results of any tests on high-temperature ATWS-like excursions that are performed in the Halden reactor. Eventually, these tests should provide data to support analyses to provide a more realistic understanding of fuel behavior under these conditions.

A plant transient code and a fuel rod code will have to describe dryout and rewet (and the eventual failure to rewet) during the oscillations along with cladding temperature and oxidation. This may require a significant upgrade of dryout and rewet models for oxidized fuel cladding in TRACE, but the combined TRACE and PARCS plant-system codes should then be able to calculate the reactor conditions.[*] Calculations of local oxidation and cladding temperature can be done with FRAPTRAN, and this will also require further developments. The initial FRAPTRAN code was completed in 2001, but it does not contain adequate cladding-to-coolant heat transfer models for axially dependent dryout and rewet conditions. Cooperative work with the Radiation and Nuclear Safety Authority (STUK) in Finland appears to offer some of the improvements that are needed. The Finnish single-channel GENFLO thermal-hydraulics code was coupled with FRAPTRAN by the Finns and was recently installed on our contractor's computers. This combination of codes will be used in the interim as further code improvements are being made in TRACE and FRAPTRAN. Calculations of peak cladding temperatures and total cladding oxidation are being planned at this time, and the results will be compared with the same criteria that are used for LOCA analysis.

Because the resolution of the ATWS issue may involve LOCA-like cladding embrittlement criteria, this issue cannot be resolved until the LOCA criteria are confirmed or modified for high-burnup BWR fuel. For BWRs, which use Zircaloy-2 cladding, this is scheduled to be done by December 2004. Overall resolution of the ATWS issue, with plant calculations and confirmatory testing, might then be completed in the 2005-2006 timeframe. This work is not receiving a high priority at the present time.

(b) Review of Industry Assessment for Extended Burnup Above 62 GWd/t

The licensing strategy that was adopted in the original program plan states that the industry will address this issue. Therefore, a submittal and a corresponding review effort are expected. The staff is not aware of any industry activity on the ATWS issue at this time.

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[*]TRACE (TRAC RELAP Advanced Computational Engine) is the new name for the upgraded TRAC-M plant systems computer code.

3.5 Fuel Rod and 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, and gap activity and to perform other licensing analyses. FRAPTRAN, a transient code, is also used by NRC for special calculations and to interpret test results. Both codes can also be used to help understand fuel behavior under a range of conditions (when used along with thermal-hydraulic codes) to support risk assessments. In the late 1980s and early 1990s, when the staff was reviewing industry requests to go to 62 GWd/t, the industry was using fuel codes that had been updated for burnups in that range. However, NRC's codes had not been updated or validated for burnups above about 40 GWd/t (rod average). Thus NRC's ability to deal with high-burnup fuel issues had been hampered by outdated analytical tools.

For reactor power calculations, neither the industry nor the NRC was, as a rule, using 3-D neutronics codes during the 1980s and 1990s. Many postulated reactivity events, including 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. Although some industry 3-D codes had been submitted for NRC review, most licensing codes did not have this capability or used overly simplifying assumptions. NRC also occasionally used its own 3-D neutronics codes for special analyses, but those codes were not coupled with the NRC's principal thermal-hydraulic codes.

Development of a more dimensional neutron kinetics analysis and modification of several specific features of the kinetics codes will allow more realistic evaluation of localized high-burnup effects. For example, local power peaking during rapid power pulses (critical or prompt critical) could be modeled more realistically by codes that use fuel rod bundles as the smallest calculational node rather than individual fuel rods. The reduction in the delayed neutron fraction that results from the buildup of plutonium isotopes at very high burnups could also be assessed more realistically. These and other high-burnup code features needed to be examined carefully.

(a) Confirmatory Assessment for Burnups up to 62 GWd/t

In the original 1998 program plan, it was said that resolution of the issue of burnup capability of the codes would occur for the fuel codes when FRAPTRAN was updated to install the high-burnup thermal models that had been developed for the just-completed FRAPCON-3 code.[17] That has been done.[18] It was also said that, for the 3-D neutronics codes, resolution would be largely achieved when the new 3-D capability of the coupled TRAC-PARCS code became available. It is now available.[19] Therefore, the identified deficiencies in these codes have been corrected and this issue has been resolved.

Nevertheless, continued maintenance of and improvements in these codes are necessary in order to realistically assess and address the other issues described in previous sections. We know from experience that continued maintenance and improvement of these codes is necessary to support the realistic assessment of higher burnups and new fuel designs. In addition, the experimental programs are providing new mechanical properties data for cladding on high-burnup fuel.

Burnup extensions from 62 to 75 GWd/t will probably be requested first for PWRs, which have traditionally operated at higher burnups than BWRs. It is expected that such burnup extensions will be done not with Zircaloy cladding, but with the advanced cladding alloys, ZIRLO and M5. Therefore, improvements in NRC's fuel codes for applications to 75 GWd/t burnup will depend on the availability of cladding properties data for ZIRLO and M5 from the Argonne program, which was discussed in Section 3.3. These data are expected to become available in 2006 for ZIRLO and 2008 for M5, depending on the availability of high-burnup specimens.

(b) Review of Industry Assessment for Extended Burnup Above 62 GWd/t

To perform the safety analyses needed for burnup extensions to 75 GWd/t, the industry will use codes that are analogous to NRC's FRAPCON steady-state code and FRAPTRAN transient code. Some of the main industry steady-state codes are Westinghouse's PAD, Framatome's COPERNIC, and Global Nuclear Fuel's PRIME. In general, these codes have been or are being approved for burnups up to 62 GWd/t. Additional submittals and review will be required to approve these codes for use above 62 GWd/t. EPRI has also provided its transient code, FALCON, to the staff. Table 3 shows recently completed, current, and expected reviews that would be used to support industry analyses above 62 GWd/t.

Table 3. Reviews of Industry Fuel Behavior and Neutronics Codes
for Extended-Burnup Analysis

Applicant	**Description**	**Completion**
Global Nuclear Fuel	PRIME (62 GWd/t limit)	3/30/2004
Westinghouse	SP-NOVA/VIPRE (RIA Analyses)	5/30/2003 C

3.6 Source Term and Core Melt Progression

The progression of a severe accident depends 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 NUREG-1465 source term.[20] However, that report noted that the source term in the report (particularly gap activity) may not be applicable for fuel irradiated to burnup levels in excess of about 40 GWd/t. The burnup applicability limitation discussed in NUREG-1465 came from the data range of the HI and VI fission product tests at Oak Ridge National Laboratory. These tests were run for burnups up to 47 GWd/t. At higher burnups, fuel pellet microstructure changes, the gap inventory increases, and the isotopics shift. Also, cladding becomes more brittle at higher burnups, potentially affecting behavior during a severe accident.

(a) Burnups up to 62 GWd/t

In the 1998 program plan, the staff presented a rationale and concluded that it was unlikely that high burnup would have a significant effect on source terms or core melt progression. Severe accident scenarios are not expected to be significantly different at high burnup. Further, fuel management practices result in most of the fuel in the core being below 47 GWd/t burnup at

any given time. Considering this combination of factors, the staff concluded that the NUREG-1465 source term is applicable to operating reactors with maximum assembly burnups up to 62 GWd/t. This conclusion has been incorporated into Regulatory Guide 1.183.[21] Thus, the source term issue has been resolved.

(b) Burnups for Extended Burnup Above 62 GWd/t

For burnups above 62 GWd/t, the staff organized a panel of source term experts to evaluate the applicability of the NUREG-1465 source term to operating 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. The panel's assessment indicated that the revised source term is generally applicable for fuel to that higher burnup level.[22]

Nevertheless, the source term panel members recommended acquiring additional test data on fission product releases for extended-burnup fuels to confirm its assessment. Source term tests on high-burnup fuel specimens are being conducted by the French Institute for Radiological and Nuclear Safety (IRSN) in its VERCORS and PHEBUS programs and by the Japan Atomic Energy Research Institute (JAERI) in its VEGA program. The NRC will obtain the results of these high-burnup tests when they are performed. The NRC will then assess the test results.

Based on the expert panel's assessment, the staff's review, and the future assessment of IRSN and JAERI high-burnup test data, the staff may propose revisions to Regulatory Guide 1.183 in the future.

3.7 Transportation and Dry Storage

Radionuclide inventory and long-term fuel degradation are two aspects of transportation and dry storage of spent fuel that might be affected by high burnups. The radionuclide inventory affects shielding and criticality provisions in the design of storage and transportation casks, and this inventory is directly related to the level of burnup. Fuel degradation affects retrievability of the fuel, which is the subject of 10 CFR 72.122. Fuel degradation in storage or transportation casks may also be affected by burnup.

Probabilistic risk assessments (PRAs) and other studies will help risk-inform the regulations (10 CFR 71 and 72), review processes, and inspection programs in the future. A screening PRA for dry cask storage has recently been performed by the staff. The preliminary results show very low risk levels. This report is undergoing an independent peer review by Sandia National Laboratories. An update of earlier transportation studies was published in March 2000 and also showed low levels of risk.[23] Effects of severe transportation accidents will be explored as part of the Package Performance Study.[24] The industry is developing a PRA on a mechanically closed storage cask, and that PRA may complement the results of the NRC studies. However, the effects of high-burnup fuel are not explicitly considered in most of these studies, so results from work on high-burnup fuel will be assessed to confirm these risk studies.

(a) Confirmatory Assessment for High Burnups

In the 1998 program plan, issues related to transportation and dry storage were treated as subjects for future actions. Subsequently, work at Oak Ridge National Laboratory on radionuclide inventory has been used to include burnup credit in interim staff guidance for criticality safety analyses of PWR spent fuel in transportation and storage casks. Other work was initiated at Argonne National Laboratory to investigate spent fuel cladding behavior and to further measure radionuclide inventories in high-burnup fuel. Based on preliminary results, a 400 °C cladding temperature limit has been established to avoid hydrogen embrittlement of cladding during storage of spent fuel with burnup levels to 68 GWd/t (average for the peak rod). This limit is now contained in interim staff guidance documents and standard review plans for storage.

However, specific criteria on spent fuel degradation have not yet been developed for the transportation of high-burnup fuel. The staff is deferring the development of acceptance criteria for transportation until the response of high-burnup fuel cladding to transportation accidents is better understood. In the interim, the staff is reviewing spent fuel transportation applications on a case-by-case basis. In accordance with the staff's current approach to resolve the technical issues (i.e., assure cladding integrity or analyze the expected configuration of the fuel under accident conditions), additional data and analyses are needed to gain a better understanding of the geometric configuration of the spent fuel after a transportation accident. The assumption that the spent fuel cladding remains intact under accident conditions is a key assumption currently used to demonstrate compliance with the regulations with respect to the criticality, shielding, and thermal analyses.

Work is underway at Argonne National Laboratory on medium-burnup (~35 GWd/t) and high-burnup (55-65 GWd/t) 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. Preliminary results on medium-burnup fuel rods show that major hydrogen redistribution can occur under vacuum drying conditions if the temperatures or pressures are too high (see Fig. 7).

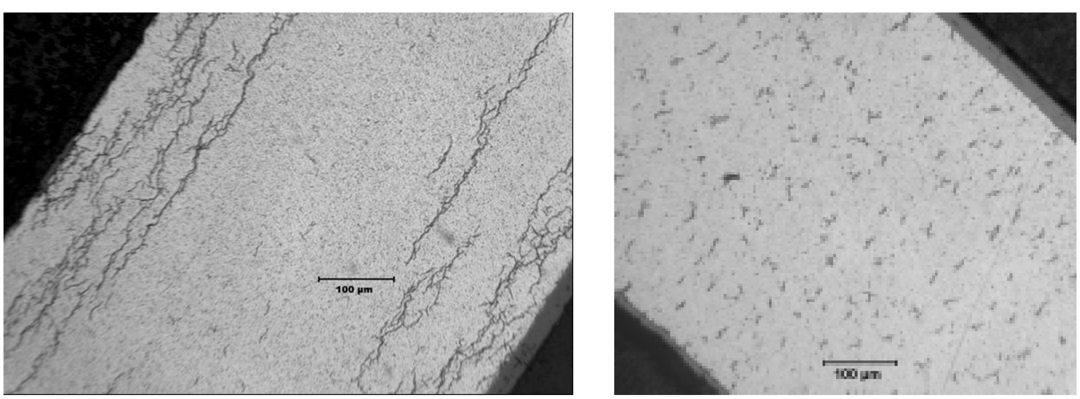

Figure 7. Normal high-burnup hydride distribution is completely reoriented after a severe simulated vacuum drying transient

The results from this research program will be used to confirm that the cladding acceptance criteria contained in the guidance documents for dry cask storage continue to provide reasonable assurance that spent fuel is retrievable from the storage cask systems — or will be used to revise the guidance documents if necessary. The results will also be used to assess the response of the cladding to high strain rate loads that the cladding might encounter in postulated transportation accidents. Results for high-burnup fuel rods with Zircaloy-2 and Zircaloy-4 cladding will be completed in 2003. Similar results for the advanced cladding alloys, ZIRLO and M5, will be obtained later when irradiated fuel rods become available for testing (see Section 3.3).

Final resolution of the technical issues associated with the transportation of high-burnup fuel with regard to burnup effects and cladding type will occur after the research at Argonne National Laboratory is completed. The NRC staff plans to use the data and results from the Argonne National Laboratory research program to (1) confirm that the acceptance criteria in existing guidance documents for dry storage are adequate and (2) develop acceptance criteria and staff guidance for conducting reviews of transportation casks. The guidance will be added to the standard review plans for storage casks and facilities (NUREG-1536 and NUREG-1567, respectively) and spent fuel transportation packages (NUREG-1617).

4 References

1. R. O. Meyer et al., "A Regulatory Assessment of Test Data for Reactivity-Initiated Accidents," *Nuclear Safety*, Vol. 37, October-December 1996, 271-288.

2. T. Nakamura, H. Sasajima, and H. Uetsuka, "NSRR RIA Tests Results and Experimental Programmes," *Proceedings of the Topical Meeting on RIA Fuel Safety Criteria*, NEA/CSNI/R(2003)8/Vol.2, April 2003, 83-95 (Aix-en-Provence conference, May 13-15, 2002)

3. J. Papin et al., "Main Outcomes from the Cabri Test Results," *Proceedings of the Topical Meeting on RIA Fuel Safety Criteria*, NEA/CSNI/R(2003)8/Vol.2, April 2003, 61-81 (Aix-en-Provence conference, May 13-15, 2002).

4. Y. Bibilashvili et al., "Study of High Burnup VVER Fuel Rods Behaviour at the BIGR Reactor Under RIA Conditions: Experimental Results," *Proceedings of the Topical Meeting on RIA Fuel Safety Criteria*, NEA/CSNI/R(2003)8/Vol.2, April 2003, 115-129 (Aix-en-Provence conference, May 13-15, 2002).

5. B. E. Boyack et al., *Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High-burnup fuel*, NUREG/CR-6742, September 2001.

6. R. O. Meyer, *Implications From the Phenomenon Identification and Ranking Tables (PIRTs) and Suggested Research Activities for High-burnup fuel*, NUREG-1749, September 2001.

7. C. Vitanza, “An Analysis of the Cabri REP Na Tests,” *Proceedings of the Topical Meeting on RIA Fuel Safety Criteria*, NEA/CSNI/R(2003)8/Vol.2, April 2003, 39-58 (Aix-en-Provence conference, May 13-15, 2002).

8. D. J. Diamond, B. P. Bromley, A. L. Aronson, “Pulse Width in a Rod Ejection Accident,” *Proceedings of the Topical Meeting on RIA Fuel Safety Criteria*, NEA/CSNI/R(2003)8/Vol.1, April 2003, 33-43 (Aix-en-Provence conference, May 13-15, 2002).

9. R. Montgomery, N. Waeckel, R. Yang, *Topical Report on Reactivity Initiated Accident: Bases for RIA Fuel and Core Coolability Criteria*, EPRI (1002865), Palo Alto, CA, June 2002.

10. Gary M. Holahan, NRC, letter to David J. Modeen, NEI, November 8, 1999.

11. B. E. Boyack et al., *Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High-burnup fuel*, NUREG/CR-6744, December 2001.

12. Ralph O. Meyer, “LOCA Ductility Tests,” *Proceedings of the 2002 Nuclear Safety Research Conference*, NUREG/CP-0180, March 2003, 99-108 (Washington conference, October 28-30, 2002).

13. V. Asmolov et al., “Understanding LOCA-Related Ductility in E110 Cladding,” *Proceedings of the 2002 Nuclear Safety Research Conference*, NUREG/CP-0180, March 2003, 109-125 (Washington conference, October 28-30, 2002).

14. *Assessment of BWR Mitigation of ATWS, Volumes I and II, (NUREG-0460 Alternate No. 3)*, NEDE-24222, December 1979.

15. *ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability*, NEDO-32047-A, June 1995.

16. B. E. Boyack et al., *Phenomenon Identification and Ranking Tables (PIRTs) for Power Oscillations Without Scram in Boiling Water Reactors Containing High-burnup fuel*, NUREG/CR-6743, September 2001.

17. D. Lanning, C. E. Beyer, C. L. Painter, *FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Application*, NUREG/CR-6534, Vol. 1, December 1997.

18. M. E. Cunningham et al., *FRAPTRAN: A Computer Code for the Transient Analysis of Oxide Fuel Rods*, NUREG/CR-6739, August 2001.

19. Thomas J. Downar et al., “PARCS: Purdue Advanced Reactor Core Simulator,” *Proceedings of PHYSOR 2002*, ANS Conference, October 7-10, 2002, Seoul, Korea.

20. L. Soffer et al., *Accident Source Terms for Light-Water Nuclear Power Plants*, NUREG-1465, February 1995.

21. William D. Travers, “Final Regulatory Guide 1.183 (Formerly DG-1081), ‘Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Plants,’ and Standard Review Plan Section 15.0.1, ‘Radiological Consequence Analyses using Alternative Source Terms’,” NRC memorandum to the Commissioners, July 19, 2000.

22. M. Khatib-Rahbar et al., *Accident Source Terms for Light-Water Nuclear Power Plants: High Burnup and Mixed-Oxide Fuels*, ERI/NRC 02-202, November 2002.

23. NRC/NMSS, *Reexamination of Spent Fuel Shipment Risk Estimates*, NUREG/CR-6672, V1, March 31, 2000 (ADAMS #ML003698324, not publicly available)

24. William D. Travers, “Spent Fuel Package Performance Study Schedule,” NRC memorandum to the Commissioners, May 2, 2002 (ADAMS #ML020990128, publicly available).