May 30, 2008

MEMORANDUM TO: Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Michael R. Johnson, Director
Office of New Reactors

FROM: Brian W. Sheron, Director ***/RA/***
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER 0801, "TECHNICAL BASIS FOR REVISION OF EMBRITTLEMENT CRITERIA IN 10 CFR 50.46"

Introduction

Emergency core cooling systems (ECCS) must be designed so that their calculated cooling performance following postulated loss-of-coolant accidents (LOCAs) conforms with specified regulatory criteria. Some of these criteria are related to embrittlement (i.e., loss of ductility) in the metal cladding on the reactor fuel rods. The current embrittlement criteria do not account for the effects of fuel burnup (the extent to which fuel is used in a reactor), and are applicable only to older cladding alloys, such as Zircaloy. This Research Information Letter (RIL) provides the technical basis to address these issues.

Background

By the mid 1990s, power reactors were achieving burnups twice as great as expected when regulations were written regarding LOCAs. Because those regulations—principally Title 10, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," of the *Code of Federal Regulations* (10 CFR 50.46) and Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"—involve fuel behavior, and because fuel behavior is known to be affected by burnup, it was suspected that those regulations might require some revision. In July 1998, the U.S. Nuclear Regulatory Commission (NRC) adopted a research plan (Executive Director for Operations (EDO) memorandum to the Commissioners, dated July 6, 1998) to address the effects of high burnup on LOCAs and other events. On August 3, 1998, the NRC issued Information Notice (IN) 98-29, "Predicted Increase in Fuel Rod Cladding Oxidation," addressing LOCAs. The IN stated that, in order to account for burnup effects, the 17-percent cladding oxidation limit in 10 CFR 50.46 should be reduced by the thickness of the corrosion layer that

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builds up during normal operation. The Office of Nuclear Regulatory Research (RES) was aware of test results for some non-LOCA transients where corrosion-related effects lead to embrittlement, and RES staff assisted in developing IN 98-29. However, the agency based the specific corrosion subtraction on judgment as no direct data were available at that time.

On March 29, 2002, the staff issued SECY-02-0057, “Update to SECY-01-0133, ‘Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria).’” In that document, the staff recommended modifications to 10 CFR 50.46 to provide for a more performance-based approach to meeting the ECCS criteria. The recommendation included research to explore post-quench ductility of cladding on high-burnup fuel and to develop embrittlement criteria that would enable licensees to use cladding materials other than Zircaloy and ZIRLO without an exemption. On March 31, 2003, the Commission issued a staff requirements memorandum (SRM) on SECY-02-0057 that approved the staff’s recommendation to modify these criteria. The staff then updated the high-burnup research plan (EDO memorandum to the Commissioners, dated August 21, 2003) to develop the technical basis for modified criteria at the earliest date possible. Sufficient research to revise the criteria has now been completed.

Summary and Significance of Results

The effects on cladding embrittlement of both alloy composition and burnup were studied in the present research program. Alloy composition effects were insignificant, but burnup effects were substantial. Some of the hydrogen that is liberated in the corrosion process enters the cladding during normal operation. The hydrogen was found to produce a strong effect on the embrittlement of the cladding, but the effect is indirect. Hydrogen acts as a catalyst while oxygen, which diffuses into the cladding metal during a LOCA transient, is the direct cause of embrittlement. Oxygen from the oxide fuel pellets was found to enter the cladding from the inside in high-burnup fuel in addition to the oxygen that enters from the oxide layer on the outside of the cladding. Under conditions that might occur during a small-break LOCA, the accumulating oxide on the surface of the cladding can break up and was found to let large amounts of hydrogen into the cladding during the LOCA transient thus exacerbating the embrittlement process. The current work also confirmed an older finding that, if rupture occurs during a LOCA transient, large amounts of hydrogen can enter the cladding from the inside near the rupture location.

Enclosure 1 discusses these major research findings in more detail. Taken together, these findings give the circumstances in which embrittlement occurs under LOCA conditions. Because the major burnup effect is caused by hydrogen, which is absorbed in the cladding during the burnup-related corrosion process, the embrittlement thresholds have been correlated directly with the preaccident hydrogen concentration. Figure 1 shows the measured embrittlement thresholds for all cladding types and burnups that have been investigated.

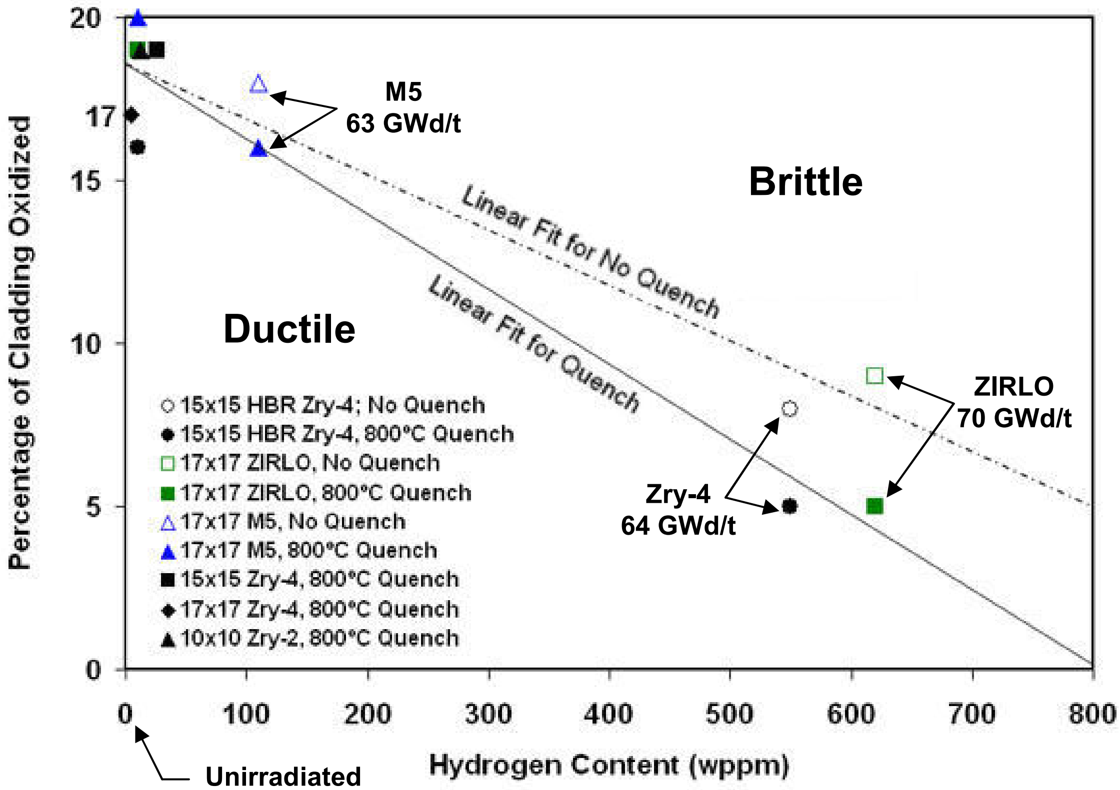

Figure 1 Measured embrittlement threshold, expressed as a percentage of cladding oxidized and calculated with the Cathcart-Pawel equation, as a function of pre-test hydrogen content

The trends seen here should apply to all present and future zirconium-alloy cladding materials because no dependence on specific alloy composition has been found. However, the adequacy of these embrittlement thresholds is subject to several conditions:

- Cladding temperatures would have to remain no higher than 1204 °C (2200 °F) because embrittlement occurs at lower oxidation values for higher temperatures.

- Calculations of cladding oxidation use the Cathcart-Pawel equation for weight gain of fresh Zircaloy because all of the data in Figure 1 were correlated with that parameter.

- Manufacturers or licensees would have to provide hydrogen-versus-burnup correlations because hydrogen absorption might vary for different materials and operating conditions.

- Some periodic testing would be needed to ensure that manufacturing processes had not changed in a way that would degrade the performance of the cladding material under LOCA conditions. Such testing could be done on as-fabricated material and would be relatively easy to conduct. Appropriate testing procedures could be defined.

- The Cathcart-Pawel equation for two-sided oxidation is used for high-burnup fuel to account for oxygen diffusion from the inside diameter of the cladding, although there would be no heat associated with a metal-water reaction on the inside diameter.

- Breakaway oxidation would have to be avoided by using an additional time limit based on tests for each cladding material. These tests could be done on as-fabricated material and would be relatively easy to perform. Appropriate testing procedures could be defined.

- The embrittlement thresholds described above would apply only to fuel rods made with zirconium-alloy cladding and containing oxide fuel pellets.

In a LOCA analysis, a reduction of fuel reactivity with burnup may offset the reduction in embrittlement threshold with hydrogen, as shown in Figure 1. However, the extent of this compensation—particularly for fuel containing burnable poisons—is beyond the scope of the experimental program described herein.

Finally, no criteria have been found that would ensure ductility in the cladding balloon. However, loss of ductility in this short portion of a fuel rod should not lead to an uncoolable geometry as long as the amount of oxidation in the ballooned region remains limited in the current manner.

The present set of data is substantially larger and more precise than the data set on which the original rule was based, and the staff of the Office of Nuclear Regulatory Research recommends that the data summarized in this RIL be considered as the basis for rulemaking to revise 10 CFR 50.46(b). Several other fuel-related LOCA phenomena are under investigation or consideration in the NRC's research programs, but they are not needed to revise this part of the rule and they are not the subject of this RIL. Enclosure 2 mentions those phenomena.

Staff members from the Office of Nuclear Reactor Regulation (NRR) and the Office of New Reactors (NRO) have reviewed this RIL and made comments that have been incorporated. Representatives from the industry and relevant international organizations have closely followed the research on which it is based. Many public presentations have been made based on this work, and the Advisory Committee on Reactor Safeguards has reviewed it several times (see Enclosure 3).

Enclosures:
As stated

- The Cathcart-Pawel equation for two-sided oxidation is used for high-burnup fuel to account for oxygen diffusion from the inside diameter of the cladding, although there would be no heat associated with a metal-water reaction on the inside diameter.

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Discussion of Research Findings

After a lengthy hearing, in 1973 the Commission described the purpose of the embrittlement criteria— they are intended to ensure that the cladding would remain sufficiently intact to retain the fuel pellets in their separate fuel rods and therefore remain in an easily coolable array (Ref. 1, p. 1095). It was understood that the fuel rods would swell and burst locally with a longitudinal split, but that the split cladding would remain in one piece if it were not too heavily oxidized and would still retain the fuel pellets. The Commission then concluded that retention of cladding ductility is the best guarantee of its remaining intact during a loss-of-coolant accident (LOCA) (Ref. 1, p. 1098).

Zirconium dioxide is brittle, and any ductility and resistance to shattering that the oxidized zirconium alloys exhibit are associated with the underlying metal. Diffusion of oxygen into the metal causes its embrittlement. Because the diffusion of oxygen is related to the oxidation process, a limit on the amount of total oxidation was established. However, temperature also affects embrittlement of the metal, so the cladding temperature was also limited. Based on data from unirradiated Zircaloy cladding, the limits to ensure cladding ductility were set at 17 percent for maximum cladding oxidation and 1204 °C (2200 °F) for peak cladding temperature.

No significant flaws have been found in the early research on unirradiated Zircaloy, and the current work has confirmed that the cladding metal underneath the surface oxide becomes embrittled rapidly above 1200 °C (~2200 °F) (Ref. 2). Therefore, no change was foreseen in the 1204 °C (2200 °F) limit, and the investigation concentrated on the cladding oxidation limit at this and lower temperatures.

One of the first discoveries in the current research was that the embrittlement threshold for a given unirradiated alloy such as Zircaloy (Zr-1.4%Sn) is not fixed at exactly 17 percent. Table 1 illustrates this by showing test results for a number of cladding materials, including three different manufacturing quantities of Zircaloy-4 that exhibit thresholds from 16 to 19 percent (Ref. 3, Section 7). The cladding types M5 and E110 are nominally the same alloy (Zr-1%Nb), yet they too exhibit very different embrittlement thresholds (Ref. 4, p. 3.59). On the other hand, zirconium alloys of different composition do not necessarily behave differently. Table 1 shows that some batches of Zircaloy-4, Zircaloy-2, ZIRLO (Zr-1%Sn-1%Nb), and M5 all have about the same embrittlement threshold. Likewise, the different alloys E110 and E635 (same nominal composition as ZIRLO) exhibit similar behavior (Ref. 4, p. 3.62). Further studies with E110 showed that variations in six major alloying elements did not affect ductility under LOCA conditions (Ref. 4, p. 5.16). Variations in embrittlement threshold are thus seen to result mainly from manufacturing differences rather than from specific alloy composition, and the starting ingot type (sponge zirconium versus electrolytic zirconium) and final surface finish on the tubing were found to be the major factors (Ref. 3, Section 3.5; and Ref. 4, p. 5.17).

Table 1 Variation of Embrittlement Threshold (Percent Equivalent Cladding Reacted, % ECR) for Unirradiated Zirconium-Alloy Cladding Materials Oxidized in Steam at 1200 °C (~2200 °F)

Alloy and Geometry	Manufacturer	Vintage	Threshold (% ECR)
Zircaloy-4, 15x15	Siemens	old, H.B. Robinson	16
Zircaloy-4, 17x17	Westinghouse	current, low tin	17
Zircaloy-4, 15x15	Areva	current, low tin	19
Zircaloy-2, 9x9	Global Nuclear Fuel	current, Zr liner	19
ZIRLO, 17x17	Westinghouse	current, standard tin	19
M5, 17x17	Areva	current	20

Tests were also performed on irradiated cladding taken from high-burnup fuel rods with Zircaloy-4, ZIRLO, and M5 cladding. As expected, the embrittlement thresholds in the irradiated materials were found at substantially lower oxidation levels, as seen in Figure 1 for the Zircaloy-4 cladding (Ref. 3, Section 5.1.2). Fresh cladding of this particular material became brittle around 16 percent, whereas irradiated cladding became brittle around 8 percent. Although oxygen diffusion into the metal is still mainly responsible for embrittlement, hydrogen that is absorbed during normal operation acts as a catalyst to accelerate the embrittlement process. Because this hydrogen comes from the dissociation of water during the corrosion process, cladding materials that corrode more readily under normal operating conditions tend to embrittle more readily under LOCA conditions. Hydrogen pickup fractions vary from one material to another, however, and the effect is characterized more precisely by the hydrogen concentration than by the amount of corrosion, as Information Notice 98-29 suggested. Testing has also shown that the cooling rate at the end of a LOCA transient somewhat alters the effectiveness of hydrogen such that the embrittlement threshold is further reduced in cladding with high levels of hydrogen if the core is quenched.

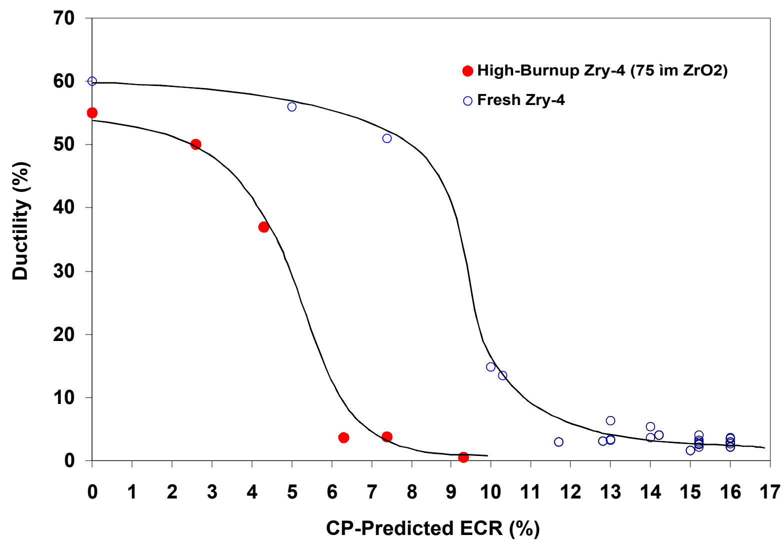

Figure 1 Ductility after oxidation in steam at 1200 °C (~2200 °F) and cooled slowly for irradiated and unirradiated Zircaloy-4 cladding; embrittlement occurs at about 2-percent ductility on this plot

Another finding for high-burnup fuel is that oxygen can diffuse into the cladding metal during a LOCA from the inside diameter (ID) as well as from the outside diameter (OD), even when no steam oxidation is occurring on the ID. The ID oxygen diffusion phenomenon was discovered in the United States in 1977, confirmed by tests in Germany in 1979, and is seen in the present results. Recent Halden tests have further confirmed it, as seen in Figure 2 (Ref. 5, Figure 3.5.24). In this figure, the thickness of the ID alpha crystallographic layer is equal to that of the OD layer, thus demonstrating that the same amount of oxygen has entered the metal from the ID and the OD. Oxygen diffusion has always been associated with ID oxidation in the balloon region where steam enters the rupture opening and reacts to form a zirconium-oxide layer. In high-burnup fuel, however, there will already be a zirconium-oxide layer on the cladding ID, even far away from the balloon, as the result of bonding between the oxide fuel pellets and the cladding. Thus, two-sided oxygen diffusion must be accounted for in high-burnup fuel to accurately predict the onset of embrittlement. It should be noted that there would be no metal-water-reaction heat associated with this process on the ID, in contrast to the situation in a rupture node.

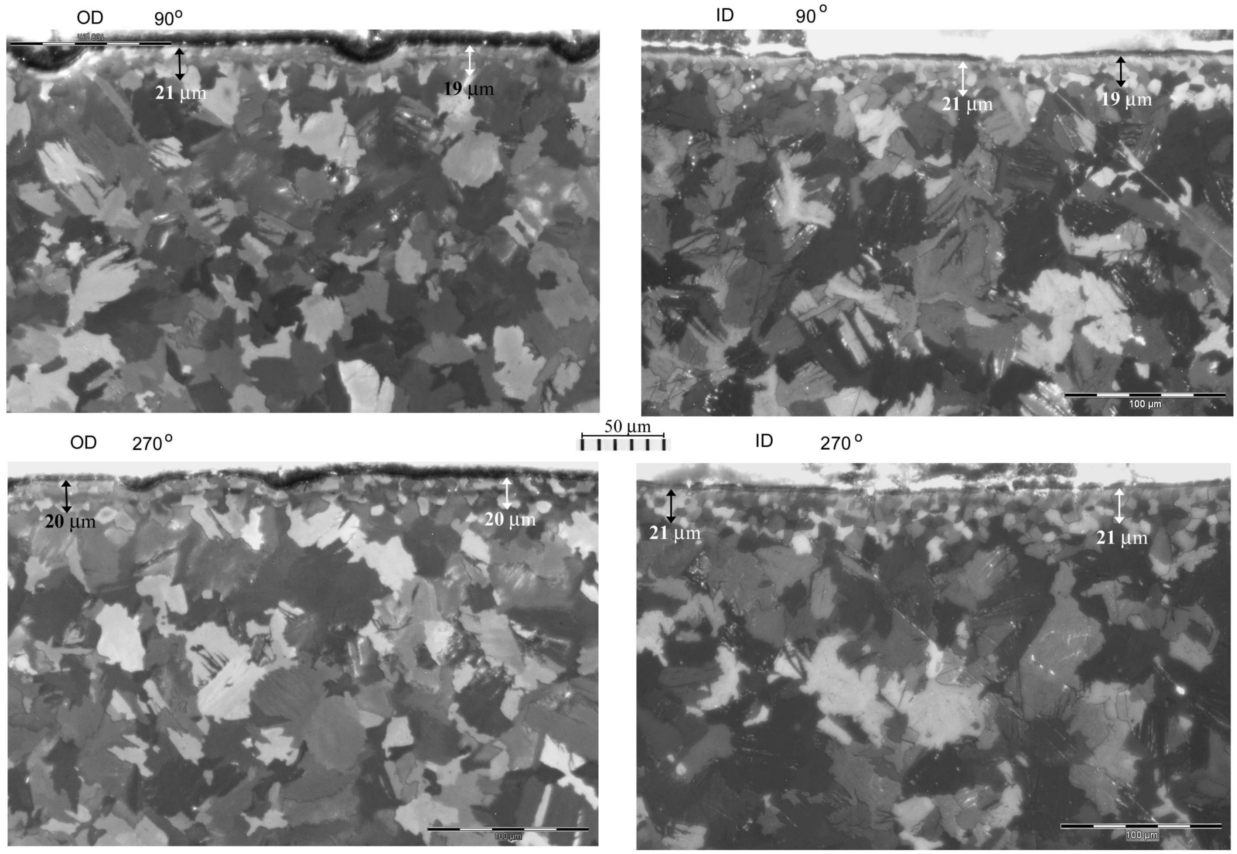

Figure 2 Micrographs from Halden test IFA-650.5 at an elevation approximately 37 cm (~14 inches) above the burst opening, showing equal ID and OD alpha-layer thicknesses (about 20 microns) at two angular locations

Many tests were also performed to determine the effect of breakaway oxidation on the embrittlement process. Breakaway oxidation occurs in all zirconium-alloy cladding materials if the cladding is held for long times (tens of minutes) at certain temperatures between 650 and 1100 °C (~1200 and ~2000 °F). Under these conditions, the crystal structure of the zirconium dioxide can change to a form that is not protective, and hydrogen from the metal-water reaction during the LOCA can then be absorbed rapidly into the cladding metal. Breakaway oxidation is often apparent from the visual appearance of the oxide layer, but a more reliable indicator is a measured increase in hydrogen concentration of 200 parts per million. Leistikow and Schanz first explored breakaway oxidation in Germany in 1987, but its degrading effect on embrittlement was not known at that time. Breakaway oxidation is also found to be sensitive to manufacturing differences, especially the surface finish. Figure 3 shows a piece of older Russian E110 cladding material that experienced breakaway oxidation on its original surface but not on a modified surface (Ref. 3, Section 3.5.2).

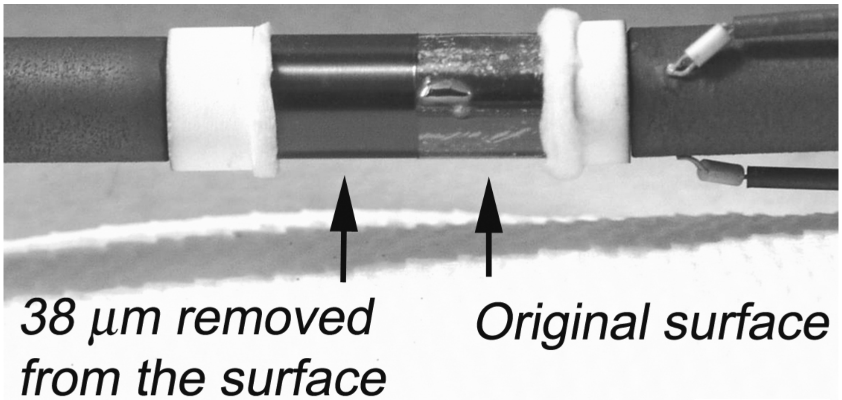

Figure 3 A piece of E110 cladding material that experienced breakaway oxidation on its original surface but not on a modified surface

Several integral tests were performed with pressurized rod segments in a simulated LOCA environment. Enhanced hydrogen absorption was observed near the cladding balloon, and this observation confirms the work of others in the 1980s. During a LOCA transient, steam enters the burst opening, and hydrogen from dissociating water is trapped inside the balloon and absorbed in the metal. This absorbed hydrogen results in rapid embrittlement just above and below the balloon. Figure 4 shows the localized hydrogen concentrations in the burst region from two tests (Ref. 3, Section 6.2.2). Loss of ductility occurs at these two peaks in hydrogen concentration. The cladding does not retain ductility at these locations, even though the oxidation in the balloon has been limited in accordance with the current regulation. Although ductility in the ballooned region cannot be assured with reasonable limits, the ballooned region has remained structurally intact during these tests with limited oxidation. Fracturing in the ballooned region has been observed during handling after the tests, but the ballooned region is generally only a few inches long, such that a fractured rod would still retain most of the fuel pellets (a major objective) as long as the fuel pellets were not too finely powdered from extremely high burnup (e.g., approximately 90 gigawatt days per ton (GWd/t), see Enclosure 2).

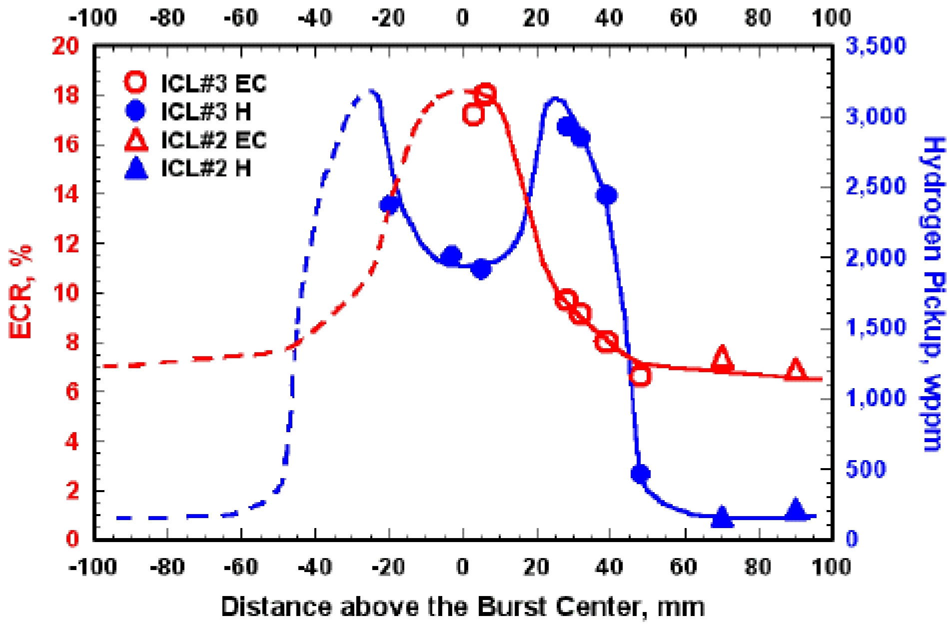

Figure 4 Axial distribution of measured hydrogen and oxygen content in the balloon region in two simulated LOCA tests (ICL#2 and ICL#3) with high-burnup boiling-water reactor fuel rods

Finally, it can be seen that none of the phenomena associated with embrittlement of the cladding metal bears any relation to the fuel pellet inside the cladding, except the diffusion of oxygen from the cladding ID. The zirconium-oxide bonding layer that supplies this oxygen develops as a consequence of oxygen migrating from the fuel pellet and combining with zirconium at the cladding ID. Because of zirconium's strong affinity for oxygen, it will pick up oxygen from any oxide in contact with it. Furthermore, because uranium, plutonium, and thorium (all potential fissile materials) have about the same free energy of formation with oxygen, there would be no significant difference in their rate of exchange of oxygen with zirconium (Ref. 3, Section 1.1). In fact, the irradiated cladding used in all tests mentioned above came from high-burnup fuel that has a substantial amount of plutonium mixed with uranium at the pellet-cladding interface. On the other hand, metal fuel would not be able to provide oxygen to the cladding ID, but metal fuel might react to form eutectics with the cladding. Thus, the cladding embrittlement processes described here would be the same for any oxide fuel pellet, but not for a metal fuel.

<u>Uncertainties</u>

All of the diffusion-related processes that affect embrittlement exhibit an Arrhenius-type exponential temperature dependence, so the largest potential source of uncertainty in research like this is the determination of temperature. The Argonne National Laboratory tests paid special attention to temperature calibration and the use of National Institutes of Standards and

Technology-calibrated thermocouple standards. The resulting report describes temperature calibration curves and temperature uncertainties in detail for each test series (Ref. 3).

Another important potential source of uncertainty in testing of this type is the limited number of duplicate tests that can be performed because of the limited number of specimens available and the difficulty of conducting each test. Although most of the test series do not include multiple tests for each set of test conditions, uncertainty can also be judged from the consistency of data along trend curves. Figure 1 above gives an example for tests with both irradiated and unirradiated samples, and the small uncertainty in these results can be seen from the small amount of scatter around the trend lines.

When applying embrittlement criteria that account for hydrogen content, an uncertainty could arise because of the axial variation of hydrogen concentration in real fuel rods. Hydrogen levels are somewhat higher opposite pellet interfaces where the cladding temperature is slightly depressed. Test results reported here employed actual measured hydrogen concentrations in the small (8-millimeter or 0.3-inch) cladding rings, which might or might not have been adjacent to pellet interfaces (pellets are about 11 millimeters or 0.4 inches long). Hence, judgment will have to be employed regarding the use of average or peak hydrogen concentrations as a function of burnup.

An additional uncertainty may arise because of the influence of cooling rate at the end of a LOCA temperature transient. Figure 1 in the cover memorandum shows the range expected for this effect.

References

1. U.S. Atomic Energy Commission, “Opinion of the Commission in the Matter of the Rulemaking Hearing on Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors,” CLI-73-39, December 28, 1973 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML993200258).

2. G. Hache and H.M. Chung, “The History of LOCA Embrittlement Criteria,” NUREG/CP-0172, May 2001, pp. 205–237 (ADAMS Accession No. ML011370559).

3. M. Billone, Y. Yan, and T. Burtseva, “Cladding Embrittlement during Postulated Loss-of-Coolant Accidents,” draft NUREG/CR-6967, May 2008 (ADAMS Accession No. ML081420276).

4 L. Yegorova, K. Lioutov, N. Jouravkova, A. Konobeev, V. Smirnov, V. Chesanov, and A. Goryachev, “Experimental Study of Embrittlement of Zr-1%Nb VVER Cladding under LOCA-Relevant Conditions,” NUREG/IA-0211, March 2005 (ADAMS Accession No. ML051100343).

5. B.C. Oberländer, M. Espeland, and H.K. Jenssen, “PIE of the High Burnup PWR Segment 650-5 after LOCA Testing in the HBWR,” IFE/KR/F-2008/064, April 2008 (ADAMS Accession No. ML081300492).

Other Loss-of-Coolant Accident Phenomena

This enclosure discusses the status of additional issues related to high-burnup fuel that the staff is addressing. However, the resolution of these issues, or lack thereof, would not affect the revision of embrittlement criteria in Title 10, Section 50.46(b), of the *Code of Federal Regulations* (10 CFR 50.46(b)).

Axial Fuel Relocation

During normal operation, oxide fuel pellets develop many cracks because of thermal stresses. Some fragmented fuel particles located above the ballooned region of a fuel rod will thus relocate into the enlarged volume of the balloon under the influence of gravity and pressure differences. This effect was first noticed in 1980 in reactor tests in the United States and Germany, and recent tests at Argonne National Laboratory (ANL) and the Halden Reactor Project in Norway have confirmed it. The consequence of fuel relocation is an increase in heat generation in the ballooned region with corresponding increases in cladding temperature and oxidation compared with an undeformed length of the fuel rod.

By 1984, the U.S. Nuclear Regulatory Commission (NRC) had classified this effect as Generic Issue (GI)-92, and it was later given a low priority based on compensating conservatisms in Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." After the best estimate option was added to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," those conservatisms were no longer guaranteed and fuel relocation was again prioritized. This time it was classified as "drop," but the evaluation that led to this classification may not have been adequate (Refs. 1 and 2). That evaluation appears to have overlooked the possibility that rapid cladding embrittlement would occur at the assumed cladding temperature of 1427 °C (2600 °F) and that embrittled fuel rods might collapse. NRC and Halden programs are performing additional testing to resolve this issue, and the resolution of GI-92 will be documented when the testing is completed.

In 2004, axial fuel relocation was the central issue in a hearing about the insertion of four lead test assemblies of mixed plutonium-uranium oxide (mixed-oxide) fuel in the Catawba plant. The contention was that axial fuel relocation would be worse in mixed-oxide fuel than in standard uranium dioxide fuel. The staff successfully argued that there would be no significant difference, but the testimony did not deny the importance of the effect itself.

Loss of Fuel Particles through a Rupture Opening

Loss of fuel particles through the rupture opening in the ballooned region of a fuel rod was not expected based on any prior research. When integral loss-of-coolant accident (LOCA) tests were recently completed at ANL on high-burnup boiling-water reactor (BWR) rods with a local burnup of 64 GWd/t, a small amount of fuel loss was noticed (about the quantity of one fuel pellet). Because the amount of material was small, this observation was not thought to be important. However, in April 2006, a LOCA test was run in the Halden reactor on a fuel rod segment with a very high local burnup of 91.5 GWd/t. Results from this test were presented to members of the project recently and showed gross loss of fuel material from above the rupture opening (Ref. 3). Online instrumentation indicated that this fuel loss occurred during the temperature transient rather than after the test was over. In this very high burnup fuel specimen, more than 40 percent of the fuel material was in a nearly powdered form referred to as a “rim.” This rim material was able to flow freely under the influence of gravity and pressure differences out the rupture opening (about 5 centimeters or 2 inches long).

Looking back at the ANL tests, the fine-grained rim region was very small compared with the Halden specimen. Consequently, the loss of a small amount of material is consistent with the loss of larger amounts of material in higher burnup fuel with correspondingly larger rim regions. The earlier German tests, however, used specimens with a maximum burnup of only 35 GWd/t, and fine-grained rim material does not appear until burnups of about 40 GWd/t. Hence, the German tests would not have been expected to lose any such fuel material through a rupture opening, and recent correspondence confirms that no fuel loss was noticed.

NRC and Halden programs are performing additional integral testing, and these tests should provide more definitive information on fuel loss during a LOCA with high-burnup fuel. However, the current NRC burnup limit of 62 GWd/t (average for the peak rod) is probably low enough to prevent significant fuel loss during a LOCA.

Ballooning and Flow Blockage

Ongoing LOCA tests at ANL and Halden are looking for differences between balloon sizes in irradiated fuel compared with unirradiated fuel. These tests are all being performed with single fuel rods rather than bundles of fuel rods. Preliminary results do not show major differences, which might have resulted from the more uniform local fuel temperatures that are expected in high-burnup fuel. This testing is not complete, however, and additional results will be analyzed carefully.

Flow blockage in pressurized-water reactor (PWR) fuel assemblies and BWR fuel bundles could occur if ballooning took place in many adjacent fuel rods. Since most rods in the hot regions of the core will balloon and rupture near their peak power elevation, such blockage is possible. In a PWR 15x15 geometry or a BWR 9x9 geometry, a diameteral strain of 67 percent on one rod will cause it to touch an undeformed adjacent rod. If both rods ballooned at that elevation, half that strain would be required for touching. Data show that, once adjacent rods touch, they do not stop ballooning, but they begin to wrap around their obstacles.

Flow blockage assumptions in most LOCA licensing analyses are based on results summarized in NUREG-0630 (Ref. 4). That report found balloon deformations to be as large as 90 percent, but they might be as small as 25 percent depending on the cladding temperature at the time of rupture. Furthermore, balloons on adjacent rods were found to occur at random axial elevations within the broad peak power region of the core. This randomness was thought to occur because of significant local temperature variations because of unevenness in the pellet-to-cladding gap. Local temperature variations set off unstable ballooning deformation at random locations.

Taking diameteral strains together with the finding of random axial rupture, NUREG-0630 found maximum flow blockages of 71 percent, but blockages could be much smaller depending on the rupture temperature. Thermal-hydraulic test data at that time had found that local flow blockages up to 90 percent do not increase cladding temperatures, largely because such blockages cause fluid turbulence, which tends to increase heat transfer. Testing above 90 percent was not performed.

A substantial amount of research on ballooning and flow blockage was conducted in the 1980s after the NRC issued NUREG-0630, yet the correlations in NUREG-0630 have not been updated. The Institute of Radiological and Nuclear Safety (IRSN) in France prepared a recent review of all these results (Ref. 5). That report claims that the 71-percent maximum blockage value is no longer supported by the experimental data and should be revised upward.

The IRSN report also points out that there have been no bundle tests with irradiated fuel rods, nor any bundle tests with modern cladding alloys. As a consequence of the IRSN concerns, a proposal was made to perform bundle tests in the IRSN PHEBUS test reactor. The proposed tests are very expensive and have not yet received widespread support.

More recently, the NRC and IRSN have undertaken formal discussions on these LOCA issues, as well as a few others. The first meeting took place at the NRC in January 2007, and followup meetings are being held. One suggestion under consideration is to perform parametric calculations with postulated flow blockages to determine potential changes before proceeding with an expensive bundle test program. Office of Nuclear Reactor Regulation, Office of New Reactors, and Office of Nuclear Regulatory Research staff are involved in these discussions, and appropriate actions will be taken based on these discussions and results from the ongoing tests mentioned above.

References

1. R. Emrit, R. Riggs, W. Milstead, J. Pittman, and H. Vandermolen, “A Prioritization of Generic Safety Issues,” NUREG-0933, Section 3, Issue 92, “Fuel Crumbling During LOCA” (Rev. 1), October 2006 (Electronic Reading Room at http://www.nrc.gov).

2. R. Meyer, NRC memorandum to J. Flack, “Update on Generic Issue 92: Fuel Crumbling During LOCA,” February 8, 2001 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML010390163).

3. W. Wiesenack and L. Kekkonen, “Overview of Recent and Planned Halden Reactor Project LOCA Experiments,” Presentation at the Japan Atomic Energy Agency’s Fuel Safety Research Meeting, May 16–17, 2007 (available from the Institute for Energy Technology (IFE)/Halden site at (http://www.ife.no/publications/2007/nusp/publication.2007-06-05.4203051771) or via ADAMS Accession No. ML081370546).

4. D. A. Powers and R. O. Meyer, “Cladding, Swelling, and Rupture Models for LOCA Analysis,” NUREG-0630, 1980

5. C. Grandjean, “A State-of-the-Art Review of Past Programs Devoted to Fuel Behavior under LOCA Conditions. Part One. Clad Swelling and Rupture. Assembly Flow Blockage,” IRSN Report, NT SEMCA 2005-313, December 2005 (ADAMS Accession No. ML072040282).

Comments of the Advisory Committee on Reactor Safeguards

The last two meetings with the Advisory Committee on Reactor Safeguards (ACRS) on the embrittlement criteria in Title 10, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," of the *Code of Federal Regulations* (10 CFR 50.46) took place on September 8, 2005, and February 2, 2007. A more detailed meeting with the ACRS Subcommittee on Materials, Metallurgy, and Reactor Fuels preceded each event. Proposed embrittlement criteria were described at both meetings.

ACRS Meeting of September 8, 2005

No letter was requested from ACRS at the first meeting, but on December 8, 2005, ACRS commented on this work in a meeting with the Commission. The following excerpt from the transcript of that meeting summarizes the ACRS comments to the Commission:

> The criteria they have developed are fairly clever and well researched. They do result in having an alloy independent process for assessing cladding, but there are still very much technology-specific regulations.
>
> ACRS reviewed this work. Our conclusions are an excellent piece of research, a well-conducted exemplary piece of research in that it involved a substantial coordination between not only NRC researchers, but industry researchers as well.
>
> We agreed that it was very important to update the regulatory requirements. In fact, these particular requirements in 50.46 have needed updating for 20 years.
>
> The regulatory requirement, however, we thought should be to preserve core coolability in design-basis accidents. That any technology-specific requirements ought to be relegated to the regulatory guides associated with that technology-specific application.
>
> This would have the effect of making this particular aspect to the regulations technology neutral and still preserve the detailed work that the staff had done in researching this particular high burnup clad behavior.

ACRS Meeting of February 2, 2007

A letter was requested from the ACRS at the second meeting, and that letter contained the following three similar recommendations:

(1) The staff should revise the embrittlement criteria in 10 CFR 50.46(a) and (b) to be technology neutral and focus on the preservation of core coolability following actuation of the emergency core cooling system in a design-basis loss-of-coolant accident (LOCA).

(2) Regulatory guides specific to zirconium-alloy-clad oxide fuels should describe acceptable methodologies to analyze fuel and cladding behavior during a LOCA.

(3) The staff should complete planned research needed to provide a sound basis for a regulatory guide applicable to current and future zirconium-based cladding alloys.

In response to the first two recommendations, the staff agreed that the U.S. Nuclear Regulatory Commission (NRC) should strive to develop performance-based requirements for light-water reactors (LWRs) for inclusion in 10 CFR 50.46 that are not based upon any specific fuel type or cladding material. The requirements would be aimed at ensuring that fuel maintains adequate structural integrity during a LOCA so that coolable geometry and long-term cooling capability are maintained and accident conditions do not challenge containment integrity. The staff also agreed that to support a performance-based rulemaking, the NRC must develop detailed regulatory guidance on acceptable methodologies for evaluating different types of fuel and cladding behavior to demonstrate compliance with the requirements.

However, an important regulatory challenge in developing a performance-based regulatory requirement for any type of fuel cladding is the need to develop objective and legally enforceable acceptance criteria. Since regulatory guidance is not enforceable, it cannot substitute for regulatory requirements.

Thus, if the staff cannot develop an enforceable performance-based regulation independent of fuel type, the staff would have to develop performance requirements for specific fuel types, but with the goal of maintaining as much flexibility as possible.

The existing 10 CFR 50.46(a) regulation is currently applicable to uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding. Modifying the NRC’s cladding requirements to be technology neutral (i.e., independent of LWR fuel type and cladding material) would involve changes to other parts of 10 CFR Chapter I in addition to the acceptance criteria specified in 10 CFR 50.46(b), such as a restriction of the option in Appendix K, “ECCS Evaluation Models,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” to only those cladding types for which the licensee can demonstrate the applicability of the Appendix K correlations.

In response to the third recommendation, the staff agreed with ACRS that planned research on zirconium-based fuel cladding should be completed. Since the staff’s presentation to ACRS in February 2007, significant additional testing has been completed on the subjects of breakaway oxidation, quench temperatures, and cooling rates. This testing also includes embrittlement measurements on irradiated ZIRLO and M5 cladding. These final tests have now been completed and are described in the final report, NUREG/CR-6967.

Consistent with the high-burnup fuel research plan, other effects of fuel behavior under LOCA conditions, such as balloon size and fuel particle relocation, will be addressed under the second phase of the research plan. Its completion is contingent on obtaining an adequate supply of irradiated fuel rods from the industry. The staff is working with the industry to obtain these samples. These other effects do not impact the zirconium cladding embrittlement criteria needed for development of the rulemaking.

References

1. “United States of America Nuclear Regulatory Commission Meeting with the Advisory Committee on Reactor Safeguards (ACRS), Rockville, Maryland, Thursday, December 8, 2005,” transcript, p. 14 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML053470458).

2. M. Bonaca (ACRS Acting Chairman) letter to D. Klein (NRC Chairman), “Proposed Technical Basis for the Revision to 10 CFR 50.46 LOCA Embrittlement Criteria for Fuel Cladding Materials,” May 22, 2007 (ADAMS Accession No. ML071430639).

3. L. Reyes (NRC Executive Director for Operations) letter to M. Bonaca (ACRS Acting Chairman), “Proposed Technical Basis for the Revision to 10 CFR 50.46 Embrittlement Criteria for Fuel Cladding Materials,” July 11, 2007 (ADAMS Accession No. ML071640115).