An Assessment of Fuel Damage in Postulated Reactivity-Initiated Accidents

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for Operating Reactors in the U.S.

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Abstract

In late 1993 and early 1994, tests in France and Japan showed that damage in fuel rods with burnups above 50 GWd/t occurs at much lower energies than in unirradiated fuel rods when exposed to large power pulses. During the last decade, significant additional test results have become available to permit an interim assessment of reactivity-initiated accidents in reactors with fuel burnups above 40 GWd/t, which is generally regarded as high-burnup fuel. These data are summarized, and systematic biases due to atypical test conditions are identified. The magnitude of biases in the fuel enthalpy for failure are estimated to range from -19 to +27 cal/g for the cases analyzed. With these adjustments, a lower bound of the experimental data is compared with potential power pulses in PWRs and BWRs. Based on available information on control rod worths, it is concluded that current operating reactors in the U.S. are not likely to experience cladding failure during the worst postulated reactivity-initiated accidents.
In late 1993 and early 1994, tests in France and Japan showed that damage in fuel rods with burnups above 50 GWd/t occurs at much lower energies than in unirradiated fuel rods when exposed to large power pulses. These pulses were intended to simulate power transients in postulated reactor accidents. The accidents being investigated were a PWR rod-ejection accident and a BWR rod-drop accident, which together are referred to as reactivity-initiated accidents (RIAs). During the last decade, significant additional test results have become available to permit a quantitative assessment of RIAs in reactors with fuel burnups above 40 GWd/t, which is generally regarded as high-burnup fuel. These data are summarized in Section 1, systematic biases in the data are estimated in Section 2, and cladding failure levels based on corrected data are compared with potential RIA pulses in PWRs and BWRs in Section 3 to determine if fuel damage would occur during these accidents.

1. SUMMARY OF RIA TEST RESULTS WITH IRRADIATED FUEL

Most of the early RIA tests were performed in the U.S. with unirradiated specimens and fuel rods with low burnup. Most of the testing of high-burnup fuel rods has been done in recent years in France, Japan, and Russia. All of these results for fuel rods with burnup are summarized here.

SPERT

The earliest RIA tests with irradiated fuel rods were performed during 1969 and 1970 in the SPERT test reactor for the U.S. Atomic Energy Commission. There were several SPERT facilities with different cores, and the core used for the tests of interest was the Capsule Driver Core (CDC), hence these tests are often referred to as CDC tests. Single rods were tested in an instrumented water-filled capsule at ambient conditions. SPERT with the Capsule Driver Core had a natural pulse width of about 20 msec.

Table 1 lists the characteristics of the irradiated fuel tests in the SPERT reactor. The test rods were BWR-type fuel rods manufactured to specifications being used by General Electric
Co. at that time, except that one group of rods had a smaller outside diameter in order to achieve higher energy depositions. These smaller rods also had a correspondingly reduced cladding thickness. Preirradiation to accumulate the burnup was done in the Engineering Test Reactor (ETR). For test CDC-756 and -859, cladding surface oxide thicknesses before the RIA test were estimated from the increase in rod diameter. If the difference in densities between metal and oxide had been properly accounted for (Pilling-Bedworth ratio), the calculated oxide thicknesses would have been larger than the reported 65 microns; however, this 65-micron oxide thickness is larger than expected from FRAPCON calculations based on ETR conditions. Although the uncertainty in this value is therefore large, results from these two tests have only a small influence on the results reported here.

**PBF**

During the period 1978-1980, RIA tests were performed in the PBF test reactor for the U.S. Nuclear Regulatory Commission. The reactor consisted of a driver core in a water pool and a pressurized water test loop that could provide a wide range of test conditions. PBF had a natural pulse width of about 15 msec.

Table 2 lists the characteristics of the fuel tests in the PBF reactor. Tests ST-1 through ST-4 were single-rod tests with fresh PWR-type fuel rods. The remainder of the tests were performed with PWR fuel rods from the Saxton PWR prototype reactor. Tests RIA 1-1 (rods 801) and RIA 1-2 (rods 802) each contained four fuel rods, but they were in individual flow shrouds such that they behaved as single-rod tests. Test RIA 1-4 (rods 804) was a true multi-rod test with a 3x3 array of nine fuel rods. Test energies were relatively high in the PBF test series because that program was designed to examine fuel behavior near the melting temperature of UO$_2$.

**Cabri**
The first test with high-burnup fuel that exhibited cladding failure at a low fuel enthalpy was performed in the Cabri test reactor, which was operated by the Nuclear Safety and Protection Institute in France (IPSN, now IRSN). The reactor consists of a driver core in a water pool and a test loop with liquid sodium for the coolant. The test loop was designed for research on liquid-metal-cooled reactors, but has been used recently to test PWR fuel with a coolant temperature of 280°C. Heat transfer in sodium is better than in water, and this is not representative of PWRs. Nevertheless, this facility is capable of energetic pulses, and it was available to test high-burnup PWR fuel specimens.

Table 3 lists the important characteristics of the high-burnup fuel tests that have been performed in the Cabri sodium loop. Although all tests with cladding failure in Cabri exhibit low fuel enthalpies at failure, the first test, REP-Na1, exhibited an unusually low failure enthalpy that has not been seen in other tests. That test has been extensively re-examined by a task force of experts, who failed to reach a consensus on the validity of the test. This author, as well as some of the task force’s experts, have concluded that the REP-Na1 test is not valid, probably as a result of hydride redistribution from preconditioning at an unusually high temperature. Task force experts from IRSN, on the other hand, think that the result is explained by extensive hydriding that is typical of reactor-spalled cladding and that the test is valid. Although the results for REP-Na1 are shown below in tables and figures, those results were not considered in reaching conclusions in this paper.

The natural pulse width for Cabri is around 9.5 msec. Based on neutronic calculations, the researchers at Cabri developed a technique to artificially broaden the pulse to be closer to expected PWR RIA conditions. Most of the tests in Table 3 were performed with broadened pulses. Pulse widths are discussed further in Section 2.
Shortly after the low energy failure occurred in the Cabri program, cladding failure with fuel dispersal was observed in a high-burnup rod at a low energy in the NSRR test reactor (test number HBO-1). The NSRR reactor is operated by the Japan Atomic Energy Research Institute (JAERI), and the reactor is a TRIGA-type annular core pulse reactor in a water pool. While a water loop has been used for some testing in NSRR, it is not used with highly radioactive irradiated fuel specimens. Tests with these specimens are conducted in an instrumented water-filled capsule at ambient conditions (standard temperature and pressure). The natural pulse width of NSRR depends on the inserted reactivity and is 4.4 ms when the largest pulse is used.

Tests on medium-to-high-burnup fuel have been underway in NSRR since 1989, and much of the early work on medium-burnup fuel rods was described earlier. Table 4 lists the important characteristics of the more recent high-burnup fuel tests with commercial PWR fuel rods. Table 5 lists the important characteristics of medium-burnup fuel tests with PWR-type fuel rods that were preirradiated in the Japan Materials Test Reactor (JMTR) or the Advanced Thermal Reactor (ATR) and then tested in the NSRR test reactor. Table 6 lists characteristics of medium-burnup and high-burnup fuel tests that were performed with fuel rods from commercial BWRs in Japan. Some values in Tables 4-6 have been updated based on recent private communications from JAERI.

**IGR**

During the 1980s and early 1990s, a large series of reactivity-accident tests was carried out in the IGR test reactor by the Russian Research Center "Kurchatov Institute." The IGR reactor is a uranium-graphite pulse reactor with a central experimental channel. Tests were performed with specimens in capsules under ambient conditions. As a rule, an experimental capsule contained two fuel rods: one high-burnup fuel rod and one fresh fuel rod. For safety reasons, instrument penetrations were not used when irradiated specimens were being tested, so
the tests with high-burnup fuel were not instrumented. The natural pulse width for this reactor was about 700 msec, which is much broader than the pulses mentioned above.

Table 7 lists the characteristics of the high-burnup fuel tests in the IGR reactor.\textsuperscript{19,20,21} These tests were performed with fuel rods from a commercial VVER in Russia. The main difference between the VVER fuel rods and PWR fuel rods is that the VVER rods have a different cladding alloy and a centerline hole in the fuel pellets.

**BIGR**

As more recent tests were being conducted in Cabri and NSRR, it was realized that pulse width could have an important impact on test results. Because IGR had such a large pulse width, additional tests that were similar to those in IGR were conducted in the BIGR test reactor.\textsuperscript{22,23} The BIGR reactor has a very narrow pulse width of about 3 ms, in sharp contrast to the IGR reactor. Table 8 lists the characteristics of the high-burnup fuel tests in the BIGR reactor.

**Graphical Display of Data from Test Reactors**

Peak fuel enthalpy is the usual metric for fuel damage in the analysis of reactivity accidents. Thus, peak fuel enthalpy values for all of the tests mentioned above are shown as a function of burnup in Fig. 1. Open symbols indicate tests without cladding failure, and solid symbols indicate tests with cladding failure. Two things are clear from this plot. First, there is a general downward trend in peak fuel enthalpy for cladding failures as burnup increases. Second, there is a lot of scatter in the data as plotted in this figure.

**2. ESTIMATES OF SYSTEMATIC BIASES IN THE DATA**

Not all of the scatter in Fig. 1 is random. Some of the scatter results from the choice of parameters on the abscissa and ordinate, and some of the scatter is a consequence of test conditions that were different from LWR conditions that would exist during an RIA. Using an understanding of these factors, substantial improvements have been made in the interpretation of these data.
Mechanisms of Cladding Failure and Fuel Dispersal

Early tests in TREAT and SPERT used fuel specimens with low or zero burnup. MacDonald reported that consequences of unirradiated fuel rod failure were insignificant at a radial average peak fuel enthalpy less than about 240 cal/g, whereas prompt fuel dispersal was observed above 275 cal/g.¹ Coolant pressure pulses were also observed at the higher enthalpies. This behavior is consistent with the melting of UO₂. Melting in unirradiated UO₂ begins at 267 cal/g (the solidus temperature), and there is a volumetric expansion of about 10 percent when UO₂ changes from the solid phase to the liquid phase. This sudden and large expansion leads to cladding failure and prompt fuel dispersal, which in turn can cause energetic fuel-coolant interactions. MacDonald also observed that previously irradiated rods would experience cladding failure at much lower fuel enthalpies, and he referred to this mechanism as a pellet-cladding mechanical interaction (PCMI). Pellet expansion, which drives this PCMI failure mechanism, is largely the result of thermal expansion, although expanding gas has also been considered as a contributor. Released fission gas, even at high burnups, produces a pressure that is less than a few percent of the cladding’s yield strength, so its contribution is not significant. Dynamic expansion of unreleased fission gas that resides in small bubbles has also been considered as a contributor to the cladding stress. Such bubbles are gas-filled UO₂ voids, and void growth (transient swelling) would require the diffusion of vacancies, which probably cannot occur during the millisecond time scale of an RIA pulse. Without transient swelling, the high gas pressure in the small bubbles cannot be transmitted to the cladding, and this effect is probably insignificant as well.

Failures from this pellet-cladding mechanical interaction (PCMI) are even more prevalent at high burnups (i.e., greater than about 40 GWd/t) because of lower cladding ductility. Enhanced corrosion of cladding at high burnup leads to significant hydrogen absorption. For each molecule of water that reacts with zirconium to form an oxide in the corrosion process, two atoms of
hydrogen are released. About 15% of this hydrogen is absorbed in the zirconium. Only about 50 to 100 ppm of hydrogen are soluble in Zircaloy at 280-330°C, and above those concentrations the excess hydrogen precipitates as zirconium hydride. Concentrations of 200-800 ppm are common in some zirconium alloys at fuel burnups around 60 GWd/t, so most of their hydrogen will be in the form of hydride precipitates. Zirconium alloys containing high concentrations of hydride precipitates are more brittle than un-hydrided metal, especially for temperatures below 400°C, and cracks initiate easily. As a consequence, the cladding cannot always deform sufficiently to accommodate the thermal expansion of the fuel pellet, even when fuel melting does not take place, and through-wall cracks develop.

However, some zirconium alloys containing niobium have corrosion levels that are so low on high-burnup fuel that the cladding has a lot of ductility, and cladding failure does not occur by the PCMI mechanism at a low enthalpy level. This was the case for the IGR and BIGR tests with the Russian E110 cladding alloy. Therefore, it is clear that burnup is not the dominant variable with regard to cladding failure during an RIA pulse. Cladding failure is seen to be much more strongly dependent on corrosion (oxidation) than on burnup. Consequently, the data have been re-plotted as a function of oxide (corrosion) thickness in Fig. 2.

Two additional features of Fig. 2 are significant. First, the stress that the expanding pellet applies to the cladding is more closely related to the change in fuel enthalpy during the test (or an RIA) than to the peak fuel enthalpy, because some stress relaxation will occur during preconditioning at the test temperature. Thus the change in fuel enthalpy has been plotted rather than the total fuel enthalpy. Second, many of these tests were instrumented such that it was possible to tell when the failure occurred and therefore to determine the actual fuel enthalpy change at the time of failure. Thus, the failure points (solid symbols) show the fuel enthalpy change at the time of failure when it is known (see Tables 1-8).
The downward trend of failures is still present, but data scatter in Fig. 2 has been substantially reduced compared with Fig. 1. Some of the remaining scatter is present because test conditions were not the same from test to test, and the most notable differences are in test temperature and pulse width. These test conditions have not only been varied, but they are also not always similar to PWR or BWR accident conditions. Variations and atypicalities in pulse width and test temperature are addressed by scaling.

**Pulse Widths and Cladding Temperature**

During a postulated rod-ejection accident in a PWR, the rate of reactivity insertion would be high enough for the reactor to go prompt critical. The resulting linear power might be many times higher than during normal operation, and the shape of the power pulse is determined by the reactivity inserted, Doppler feedback, and other factors. An inverse relation exists between pulse height and pulse width, and this relation is given analytically in an ideal case by the Nordheim-Fuchs equation.\textsuperscript{25} We have explored this relation for several real reactor cases, and we have examined calculations performed by others.\textsuperscript{26} From these calculations, it is found that pulse width (full width at half maximum) as a function of the maximum change in fuel enthalpy during a PWR transient varies as shown in Fig. 3.

The main consequence of pulse width during an RIA is that narrow pulses are more adiabatic than broad pulses, and this affects cladding temperature. Thus, a given cladding stress or strain occurs at a higher cladding temperature with a broader pulse. For the broader pulse, the higher cladding temperature may, in turn, reduce the tendency for cladding failure.

During a postulated rod-drop accident in a BWR, the rate of reactivity insertion is slower than in a PWR because of velocity limiters on each control blade. For this and other reasons, RIA pulses in a BWR are generally broader than in a PWR.

**Outline of Scaling Method**
Most of the test data for fuel with corrosion levels of interest have been obtained with PWR fuel rods in the Cabri test reactor in France and the Nuclear Safety Research Reactor (NSRR) in Japan. However, neither of these facilities reproduces the conditions of a commercial reactor (pulse shapes, fuel rod and coolant pressures, heat transfer, and in some cases temperature), so it is desirable to scale these data to PWR conditions to obtain relevant results.

For a test with cladding failure, a calculation was first performed with the FRAPTRAN fuel rod transient code for the conditions of the test, including the exact shape of the power pulse used in the test. Code predictions of cladding stress and strain at the measured time of failure were taken to be the failure stress and failure strain for the fuel rod segment being tested. A second calculation was then performed for the same fuel rod segment, but with PWR conditions. This involved using a different pulse shape, fuel rod pressure, coolant pressure, and heat transfer coefficient, all of which were appropriate for a PWR. From the second calculation, fuel enthalpy and other parameters were obtained at the time that the calculated cladding stress or strain reached the failure values that had been deduced from the first calculation. Cladding temperatures were not always the same in both calculations, so an adjustment to the deduced failure values had to be considered to account for any temperature dependence of ductility and toughness. The details of using a failure stress or a failure strain and their temperature dependencies are discussed in the following sections.

Notice that the absolute accuracy of the deduced failure stress and failure strain are only of secondary importance. Any errors introduced in the test pulse calculation will be repeated in the PWR calculation because the same code is being used, and the errors will largely cancel out.

Three tests in Cabri that resulted in cladding failure (test REP-Na1 excluded) were analyzed. In addition, two tests with cladding failure in different NSRR test series were analyzed. Analysis of these five test results – together with the data in Fig. 2 – provide an estimate of the failure enthalpies for cladding under PWR conditions.
In the paragraphs that follow, reference is made to elastic and plastic properties of the cladding. The terms that are used are defined in Fig. 4 in relation to a typical engineering stress-versus-strain plot.

**Variation of Failure Stress with Temperature**

For cladding failures that occur in the elastic region, hoop stress is the most important failure parameter. Failure is assumed to occur at a specific stress (the failure stress), which is a function of the fracture toughness, flaw size, and a geometric constant representative of the configuration of the cracked structure. However, for scaling purposes, the flaw size, distribution of flaws, and thus the geometric constant are assumed to be identical for a given specimen under test reactor and commercial reactor conditions; therefore, fracture toughness is the only property that would vary with temperature.

Fracture toughness of Zircaloy-4 containing hydrogen is low at low temperatures, but by 250°C the fracture toughness starts increasing rapidly. Although the exact temperature of this large increase may vary with hydrogen content and other parameters, there is a transition from brittle to ductile failure somewhere above 300°C.

Scaling of test data using hoop stress as the key parameter was performed first for the Cabri test, REP-Na10, because its pulse shape was less distorted than the others. The cladding in this test failed with little or no change in cladding diameter. This test, like all tests in Cabri, was conducted with an initial uniform temperature of 280°C, which corresponds to the hot zero-power temperature in a PWR. The average cladding temperature increased almost 200 degrees by the time of reported cladding failure, according to our calculations. By this time, the cladding average temperature was well above the brittle-to-ductile transition temperature, yet macroscopically ductile failure (i.e., significant diameter change) was not observed. The implication is that temperature-related diffusion processes that alter the fracture toughness were not able to operate sufficiently during the short time of the pulse to make the cladding ductile.
Therefore, in the analysis that follows, it is assumed that fracture toughness is frozen at the initial test temperature and does not vary during the pulse.

**Variation of Failure Strain with Temperature**

For cladding failures that occur with some plastic hoop strain, ductility is the most important failure parameter. For these types of failures, strength may no longer be significant as a failure limit because cladding deformation will progress beyond yielding (yield strength) – and beyond the maximum stress value (ultimate tensile strength) in some cases. Thus a plastic strain was taken as the failure parameter.

Plastic deformation depends on the number of lattice defects (including radiation-induced defects) among other factors, and the concentration of these defects can be temperature dependent. However, if the temperature changes, some time is needed for the concentration to change because this requires atomic migration (diffusion). The need for some elapsed time would show up in tests as a heating rate effect, and there are data that show such a reduction in temperature dependence with increasing heating rate.\(^{30}\) Therefore, for the very rapid heating that occurs during RIA pulses, the assumption is made that the plastic properties are frozen at the initial test temperature and do not vary during the pulse — just as was assumed for fracture toughness.

Because the Cabri tests were all performed with an initial temperature of 280°C, the appropriate temperature for a hot zero-power PWR rod ejection accident, no adjustment was needed for the temperature variation of the failure strain in those tests. However, the NSRR tests were conducted with an initial temperature near room temperature, and an explicit temperature adjustment was considered for those tests.

For failures with plastic deformation, failure can occur by propagation of a mixed-mode crack (brittle and plastic) or by local plastic instability (necking). Therefore, it is not clear whether the failure strain would be a uniform elongation or a total elongation (i.e., a non-uniform
deformation). Unfortunately, the mechanical properties under RIA loading conditions are not well known and the data available show different temperature dependencies for uniform elongation (UE) and total elongation (TE).\textsuperscript{31} These data show no discernable temperature dependence of UE from room temperature to 400\textdegree C, whereas they show a strong temperature dependence of TE in that range [TE = 2 + 0.04T(\textdegree C) is a reasonable approximation].

The scaling analysis was performed with both temperature dependencies and it was found that the larger temperature dependence of TE produced (a) a scaling correction that was three times that produced by the UE assumption, (b) less consistency in the overall results, and (c) a less conservative result. Because the more conservative result of the UE assumption is sufficient to reach the conclusion of this assessment and because codes like FRAPTRAN only calculate uniform deformation, the temperature dependence of UE (i.e., nil) was chosen for the rest of the analysis.

**Variation of Elastic Properties with Temperature**

Young’s modulus and yield points for the cladding are built into the FRAPTRAN code and they change with temperature. Figure 5 shows the values in FRAPTRAN for two different temperatures. Notice that for any given strain value, the stress is higher at the lower temperature. Because strain is related to the fuel pellet’s thermal expansion, and hence fuel enthalpy, this shows in general that higher stresses are generated at lower temperatures for the same fuel enthalpy. Conversely, yielding always occurs at a smaller strain value (i.e., limit strain) at higher temperatures.

Whereas the toughness and plastic properties discussed above involve movement of lattice defects and require time to change, the elastic properties are determined by interatomic forces acting in a perfect lattice and will respond instantaneously. Consequently, the elastic properties change with temperature during the RIA pulses and affect the results even though we
have assumed that plasticity and toughness do not change during the pulses. These changes are accounted for in FRAPTRAN.

**Strain Data and Code Assumptions**

Measured plastic hoop strains for tests with non-failed cladding in Cabri and NSRR are shown in Fig. 6. Strain was not measured for tests with cladding failure. Strains above 3%, which were observed at high energies in some tests, have been omitted from this figure because those large strains are the result of high-temperature swelling rather than PCMI and are not of interest for the purpose of this discussion. The apparent zero-strain intercepts of the various data sets in Fig. 6 correspond to the enthalpy required to close the gap and to expand the cladding through the elastic range; only then does permanent plastic deformation begin.

There is a very large spread in apparent zero-strain intercept values in Fig. 6, ranging from 25 to more than 100 cal/g. The strain data from NSRR tests appear to be segregated, with data from the PWR rods on the left in the figure, data from the BWR rods in the center, and data from rods irradiated in the JMTR and ATR materials test reactors on the right. This major grouping is believed to result from significant differences in end-of-life gap size. Hard gap closure occurs by cladding creepdown in PWR rods that have relatively thin cladding and are irradiated in the high pressure PWR environment. BWRs and low-pressure test reactors produce much less gap closure by creepdown than PWRs for a given burnup. Hence, the onset of plastic strain is delayed in the BWR rods and the rods that received their base irradiation in the test reactors.

If attention is focused on the data from PWR rods, there is still significant scatter in the NSRR data, and the NSRR data are generally to the left of the Cabri data in the figure. This relative displacement is contrary to initial expectations. Because the NSRR tests were all performed from around room temperature and the Cabri tests were all performed from 280°C, one would expect the PWR rods in NSRR to have larger initial gaps than the rods in Cabri. This would delay the onset of plastic strain more in the NSRR rods than in the Cabri rods, but the
opposite is observed. This observations shows that code-calculated initial gap sizes, which account for pellet swelling, fuel relocation, cladding creep, and other factors, will not reproduce the measured results when such gap sizes are used as input to the transient analyses. Therefore, it was necessary to override these calculated gap sizes, as described below, to get transient strain predictions that were realistic for these data sets.

Pulse width is probably not the explanation for the patterns and scatter seen in Fig. 6. The Cabri tests used a wide range of pulse widths (9-75 ms) and show very little scatter in the strain data. Incidentally, two recent Cabri tests (CIP0-1 and CIP0-2) also correlate well with the Cabri data shown, but the strain data for these tests have not been released and therefore are not plotted in the figure. The NSRR tests, on the other hand, were all conducted with about the same pulse width (~5 ms), yet those tests resulted in significant scatter in the strain data.

One possible explanation for this behavior is related to fuel chips and fines (see Fig. 7). After quiescently cooling down from its last power level in an LWR, the fuel rod’s gap will open and look something like Fig. 7(a). The gap, as shown, is probably distributed among pellet cracks as well as any open space between the pellet surface and the cladding inside surface. During specimen preparation at room temperature, it is likely that chips and fines get into the open cold gap (see Fig. 7(b)) causing lockup. This stochastic process could lead to a wide variation in effective cold gap size with regard to the stress that is applied to the cladding. The same process would take place for Cabri tests, but those specimens were subsequently held at a high temperature for a long time prior to testing. This preconditioning should compress or relocate the fuel chips and fines, leading to a more well defined effective gap, like the gap that existed in reactor. Such preconditioning is probably similar to the well known fuel conditioning that affects failure likelihood during power ramps in LWRs. Hence, the rods tested in Cabri (all PWR rods) could have a larger effective gap than the PWR rods tested in NSRR, and this would lead to a bigger delay in the onset of plastic strain in Cabri than in NSRR, as observed.
No models are present in FRAPTRAN, or any other code we know of, to account for this wide scatter in observed strain values. To compensate for this model deficiency, we manually adjusted the cold gap size to ensure that the code produced realistic strain values in agreement with measured values. For the Cabri tests, a relatively large input gap of 95 microns was found to produce reasonable agreement with measured strains. With this gap, we could then expect the code to produce approximately correct stress and strain values for the rest of the tests in Cabri (i.e., those tests with failure, for which strain was not measured). A 95 micron input gap size was thus used for the Cabri test pulse calculations and for the corresponding calculations for PWR conditions in our scaling analysis.

For the NSRR tests, a much smaller input gap size of 10 microns was needed to produce reasonable agreement with measured strains for the HBO test series. This small gap is consistent with the chips and fines hypothesis and was used for the analysis of NSRR tests. Because there is so much scatter in the NSRR data, the choice of input gap size for NSRR tests was more uncertain than for Cabri. Further, there are no NSRR test data for which the specimen was preconditioned at PWR temperatures, so the 95 micron input gap that was deduced from the Cabri tests was again used for the corresponding calculations for PWR conditions. Although these uncertainties will undoubtedly cause relatively large uncertainties in the scaling adjustments for the NSRR data, the scaling adjustments themselves are relatively small such that the overall error should also be small.

**Scaling Results**

Figures 8-12 show the actual test pulse shapes and the approximate PWR pulse shapes used in all the scaling calculations. Two important observations were made shortly after beginning these analyses. One was that the total energy in the PWR pulse had to be varied parametrically to find the smallest pulse that would produce the failure stress or failure strain that had been deduced from the actual test conditions. The other was that the deduced failure stress
or failure strain never occurred in the extreme tail of a pulse because cladding expansion eventually overtook pellet expansion. Consequently, the maximum fuel pellet enthalpy was always a little larger than the fuel pellet enthalpy at the time of failure.

In our results for PWR conditions, which are given in Table 9, the maximum change in fuel pellet enthalpy is shown for the smallest pulse that would produce failure. In our results for test pulse conditions, which are also given in Table 9, the change in fuel pellet enthalpy at the measured time of failure is shown, notwithstanding the fact that the maximum change in fuel pellet enthalpy for the test pulse was always higher (sometimes much higher). The difference between these tabulated enthalpy values gives the scaling correction, or bias, in the test data due to atypical conditions, particularly pulse width, test temperature, and excess test energy.

For plotting purposes in Fig. 13, all NSRR failure points have been adjusted to hot zero-power conditions by adding 25 cal/g, based on scaling results for HBO-1 and TK-2. The three Cabri failure points have been plotted as they were individually adjusted in our calculations. Test REP-Na1 was not plotted and has been disregarded for the reasons cited above. The Russian IGR and BIGR test data produced non-PCMI failures and have been plotted without adjustment. The PBF data were obtained with approximately the right pulse width and test temperature, and therefore were not adjusted. The SPERT data were obtained with approximately twice the right pulse width for a PWR and the tests were conducted at room temperature rather than the temperature for hot zero power. These deviations would work against each other. Nevertheless, the incompleteness of the data for these old tests did not make scaling practical, so these points were plotted without adjustment.

Some of the cladding failures in SPERT and PBF were said to occur below a reported enthalpy value, and those reported upper-bound values were plotted in Figs. 2 and 13. The fuel enthalpy at failure was not determined in any of the IGR and BIGR tests, so all of those results are plotted as maximum fuel enthalpy in the tests. On the other hand, with one exception, all of
the data points shown in Fig. 13 above 10 microns of oxide thickness are fuel enthalpy values at the time of failure. The one exception is SPERT test CDC-756, which is shown with a down-pointing arrow to indicate that the failure occurred below 143 cal/g (see Table 1). Consequently, when viewing Fig. 13, the fuel enthalpy that would produce cladding failure is defined by the lower end of the data range below 10 microns of oxide thickness and by the data points themselves above 10 microns, with the exception of the point marked with an arrow.

Although the adjusted NSRR failure points line up with other failure points in Fig. 13, these points would have moved up further in the figure had we assumed a strong temperature dependence for failure strain. In other words, the scaling adjustment for the NSRR tests is quite uncertain because of the relatively poor understanding of mechanical properties that would be applicable to RIA-type transients. The inability to determine an appropriate temperature variation from data on uniform elongation (UE) and total elongation (TE) translates to a rather large uncertainty in applying the NSRR results to hot-zero-power accidents. This conclusion would apply to a lesser degree to the SPERT results, which were not scaled for lack of detailed information and come from tests that were performed at room temperature (but with a broader pulse than NSRR).

Scaling adjustments for pulse width alone were smaller, as can be judged from the Cabri results in Table 9, and do not suffer from a major test-temperature uncertainty. Consequently, the three Cabri points in Fig. 13 are more representative of hot-zero-power accidents in PWRs and BWRs whereas the unscaled NSRR points (see Fig. 2) and the unscaled SPERT points are more representative of cold-zero-power BWR accidents.

The REP-Na8 and REP-Na10 test results, which characterize cladding failure for large oxide thicknesses, come from specimens with some oxide spallation. Spallation can lead to the formation of hydride blisters, which in principle might act as defects that could initiate cracks at lower fuel enthalpies. However, the blisters were relatively small in these specimens in relation to
the cladding thickness, and metallography does not show a significant relation between the blisters and the cracks. This observation is also consistent with recent work by Glendening et al., which concludes that thin blisters are not more effective at initiating cracks than uniform hydride rims. Therefore in the range of 80-110 microns of oxide thickness, spalling does not appear to affect the fracture process and should be thought of as merely a consequence of heavy oxide formation, which leads to an embrittled hydrided rim in the cladding.

The database contains a number of mixed uranium-plutonium oxide (MOX) fuel rods, and a MOX rod, REP-Na7, was one of the tests that was analyzed. It has been claimed that inhomogenieties in MOX fuel might affect the mechanical loading on the cladding, thus increasing the strength of the PCMI. The dynamic-gas-expansion hypothesis that would produce this increase does not appear plausible for two reasons. First, it would require rapid swelling of pores in UO$_2$, and this diffusion-controlled process is probably not fast enough for the 10-to-40 millisecond pulse periods of interest. Second, many of the plutonium-rich islands that are postulated to contribute to this effect are deep within the pellet where their expansion is overshadowed by expansion of the outer rim in these edge-peaked power excursions. Therefore, the PCMI loading, which is driven largely by thermal expansion, should be the same in MOX and UO$_2$ fuel because thermal properties of MOX and UO$_2$ are quite similar. The propensity for fuel dispersal after cladding failure might be different for MOX and UO$_2$ fuel, however.

Notice that the database in Figs. 2 and 13 contains a wide manufacturing variety of Zircaloy-2 and Zircaloy-4 cladding, as well as MDA, E110, Zirlo, and M5 cladding alloys. This latter group of alloys tends to oxidize less than Zircaloy-4, and their corrosion resistance is accounted for by correlating the failure enthalpy with oxide thickness (rather than burnup). Testing with Zirlo and M5 has been very limited to date. Nevertheless, Cabri CIP0-1 (Zirlo) provides a useful comparison that is consistent with Fig. 13. That test rod had 80 microns of oxide, experienced a 32 ms pulse, reached a peak fuel enthalpy of 90 cal/g and did not fail. This
can be compared with the Zircaloy-clad REP-Na10 test rod, which had a similar 80 microns of oxide, experienced a similar 31 ms pulse, reached a somewhat higher peak fuel enthalpy of 98 cal/g and failed. The specimen from CIP0-1 has many incipient cracks, which appear to be precursors to failure.

Considering all of the above, additional refinements and different weighting of individual data points might be possible. Nevertheless, biases from atypicalities in test conditions have been considered and it is unlikely that cladding failure would occur during hot-zero-power RIAs at fuel enthalpies below the lower bound of the points in Fig. 13.

3. COMPARISON OF CLADDING FAILURES WITH POTENTIAL RIA PULSES

In fresh or very low burnup fuel, fuel melting is needed to eject fuel from a fuel rod, and fuel melting requires high enthalpy levels. For high-burnup fuel, however, another mechanism is present to disperse fuel provided there is an opening in the cladding: the rapidly escaping fission gas that can entrain small fuel fragments. Fuel dispersal was seen in many tests with high-burnup fuel rods that had just enough energy to cause cladding failure by PCMI. Therefore, the focus of this paper is on cladding failure.

Control Rod Worth

To complete the assessment of postulated reactivity accidents in PWRs and BWRs, plant analyses had been performed to examine the conditions necessary to reach the cladding failure enthalpies observed experimentally. The PWR analysis was done with a model of the TMI-1 reactor with the PARCS 3-dimensional neutron kinetics code. Reactor conditions were considered at both end-of-cycle and beginning-of-cycle. In order to consider different control rod worths, neutron cross sections were modified for the central fuel assembly, which was the location of the ejected rod. The control rod was ejected in 100 ms from hot zero power conditions and reactor trip was initiated when power reached 112% of full power. In addition to varying control rod worth, the delayed neutron fraction at end-of-cycle was varied from 70 to
120% of the nominal value. For each transient simulation of a given rod worth and delayed neutron fraction, there was a maximum change in fuel pellet enthalpy considering all mesh points in the model.

The variation of the maximum enthalpy change with rod worth normalized by the delayed neutron fraction (i.e., in units of dollars) is given in Fig. 14. The dependence of fuel enthalpy with rod worth is approximately linear, and it increases with the delayed neutron fraction for a given rod worth in units of dollars. Although there is no comprehensive database of rod worths in U.S. power reactors, it appears unlikely for a rod worth to exceed $1.5 based on available data. For this rod worth, the enthalpy change is only 40 cal/g at beginning-of-cycle, which can be compared with cladding failures observed experimentally. This enthalpy change falls below the adjusted experimental values shown in Fig. 13, even for heavily corroded cladding. The conclusion is that cladding failure, and hence fuel dispersal, would be unlikely should an RIA occur in an operating PWR.

A similar conclusion can be reached for BWRs without performing specific analyses. Our calculations for BWRs generally result in similar values of fuel enthalpy change as for PWRs, so there should also be some margin to the BWR cold failure levels even when compared with the uncorrected NSRR failure points in Fig. 2. The broader pulse widths in BWRs may also have less of a tendency to disperse fuel when there is a cladding failure. Taken together, these factors indicate that it would be very unlikely to get cladding failure during a BWR rod drop accident.

4. SUMMARY AND CONCLUSIONS

In low-burnup fuel, fuel dispersal can be precluded by avoiding incipient melting of UO₂, which begins at 267 cal/g. By precluding fuel dispersal, energetic fuel coolant interactions cannot occur and a coolable fuel geometry is ensured. High-burnup fuel behaves differently, however. In late 1993 and early 1994, tests in France and Japan showed that cladding failure with fuel dispersal could occur at fuel enthalpies below 100 cal/g. Since that time, tests in France, Japan,
and Russia have confirmed such high-burnup behavior and generated a database that permits an updated assessment of postulated RIAs for operating reactors.

The test reactors in which the data were developed did not, however, reproduce LWR conditions well, and the atypicalities are believed to have biased some of the results. An estimate of this bias was therefore made. Using NRC’s FRAPTRAN fuel rod code, a method was developed to perform a scaling analysis to adjust raw test data for LWR conditions, and the method was used to adjust some of the most influential test results in the database. The adjustments for hot-zero-power conditions range only from -19 to +27 cal/g, so the final result is still largely empirical and closely related to the measured data. Uncertainties in the adjustments for atypical test temperatures were found to be much larger than uncertainties in the adjustments for atypical pulse widths as a consequence of the relatively poor understanding of mechanical properties that would be applicable to these very rapid transients. The adjusted data that are presented apply to Zircaloy-2, Zircaloy-4, Zirlo, and M5 cladding alloys as well as to UO₂ and MOX fuel, all of which were included in the database.

Neutronic analyses were then performed for a range of LWR conditions, and it was found that very high control rod worths (on the order of $2) were needed to deposit the amount of energy at which cladding failure was observed experimentally. There is no comprehensive database of rod worths in U.S. power reactors; however, based on available data, it is unlikely that rod worths would exceed $1.5. Therefore, it is concluded that current operating reactors in the U.S. are not likely to experience cladding failure during the worst postulated RIAs.

Finally, it can be noted that cladding failure varies only weakly with burnup level. Cladding corrosion (oxidation), which might differ widely for different cladding materials at the same burnup, was found to be the most important variable.
ACKNOWLEDGMENTS

The author would like to express appreciation to John Voglewede (NRC), Harold Scott (NRC), David Diamond (BNL), and Robert Daum (ANL) for substantial contributions to this work.
FIGURE CAPTIONS

Fig. 1. Test data, plotted as peak fuel enthalpy (total) as a function of burnup. Solid symbols indicate cladding failure.

Fig. 2. Test data, plotted as maximum fuel enthalpy change as a function of oxide (corrosion) thickness. Solid symbols indicate cladding failure.

Fig. 3. Dependence of pulse width on energy (fuel enthalpy change) for beginning of cycle (BOC) and end of cycle (EOC) conditions.

Fig. 4. Terms used to describe elastic and plastic properties of cladding.

Fig. 5. Stress versus strain in the elastic region from the elastic modulus in FRAPTRAN at two different temperatures.

Fig. 6. Plastic strain measured from non-failed cladding as a function of maximum fuel enthalpy change for tests in Cabri and NSRR.

Fig. 7. Open gap (actually, distributed cracks) after cooldown from power, (a) before handling and (b) after handling and specimen preparation.

Fig. 8. Power versus time for the broad REP-Na10 test pulse and a PWR-shaped pulse with the same deposited energy.

Fig. 9. Power versus time for the broad REP-Na8 test pulse and a PWR-shaped pulse with the same deposited energy.

Fig. 10. Power versus time for the broad REP-Na7 test pulse and a PWR-shaped pulse with the same deposited energy.

Fig. 11. Power versus time for the narrow HBO-1 test pulse and a PWR-shaped pulse with the same deposited energy.

Fig. 12. Power versus time for the narrow TK-2 test pulse and a PWR-shaped pulse with the same deposited energy.
Fig. 13. Cladding failure data with adjustments from the scaling analysis to correspond to PWR hot-zero-power conditions.

Fig. 14. Maximum fuel enthalpy change for a PWR rod ejection accident from hot zero power as a function of control rod worth for various values of delayed-neutron fraction (beta).
REFERENCES


27. K. J. Geelhood, C. E. Beyer, and M. E. Cunningham, “Modifications to FRAPTRAN to Predict Fuel Rod Failures Due to PCMI during RIA-Type Accidents," Proceedings of the


Table 1. Characteristics of BWR-type specimens with Zircaloy-2 cladding tested in stagnant water at an initial temperature of 20°C in the SPERT test reactor.*

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Test Date</th>
<th>Maximum Burnup GWd/t</th>
<th>Oxide Thickness</th>
<th>Energy Deposit cal/g</th>
<th>Pulse Width ms</th>
<th>Peak Fuel Enthalpy cal/g</th>
<th>Cladding Strain %</th>
<th>Comments**</th>
</tr>
</thead>
<tbody>
<tr>
<td>CDC-567</td>
<td>1969</td>
<td>3</td>
<td>0</td>
<td>264</td>
<td>18</td>
<td>219</td>
<td></td>
<td>PCMI failure at 214 cal/g No fuel loss</td>
</tr>
<tr>
<td>CDC-568</td>
<td>1969</td>
<td>4</td>
<td>0</td>
<td>199</td>
<td>24</td>
<td>165</td>
<td></td>
<td>PCMI failure at &lt;147 cal/g No fuel loss</td>
</tr>
<tr>
<td>CDC-569</td>
<td>1969</td>
<td>4</td>
<td>0</td>
<td>348</td>
<td>14</td>
<td>289</td>
<td></td>
<td>PCMI failure at 282 cal/g No fuel loss</td>
</tr>
<tr>
<td>CDC-571</td>
<td>1969</td>
<td>5</td>
<td>0</td>
<td>161</td>
<td>31</td>
<td>134</td>
<td></td>
<td>No failure</td>
</tr>
<tr>
<td>CDC-684</td>
<td>1970</td>
<td>13</td>
<td>0</td>
<td>200</td>
<td>20</td>
<td>166</td>
<td></td>
<td>No failure</td>
</tr>
<tr>
<td>CDC-685</td>
<td>1970</td>
<td>13</td>
<td>0</td>
<td>186</td>
<td>23</td>
<td>154</td>
<td></td>
<td>No failure</td>
</tr>
<tr>
<td>CDC-703</td>
<td>1969</td>
<td>1</td>
<td>0</td>
<td>192</td>
<td>15</td>
<td>159</td>
<td></td>
<td>No failure</td>
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<tr>
<td>CDC-709</td>
<td>1969</td>
<td>1</td>
<td>0</td>
<td>238</td>
<td>13</td>
<td>198</td>
<td></td>
<td>PCMI failure at 190 cal/g No fuel loss</td>
</tr>
<tr>
<td>CDC-756</td>
<td>1970</td>
<td>32</td>
<td>65</td>
<td>176</td>
<td>17</td>
<td>146</td>
<td></td>
<td>PCMI failure at &lt;143 cal/g No fuel loss</td>
</tr>
<tr>
<td>CDC-859</td>
<td>1970</td>
<td>32</td>
<td>65</td>
<td>190</td>
<td>16</td>
<td>158</td>
<td></td>
<td>PCMI failure at 85 cal/g Very little fuel loss</td>
</tr>
</tbody>
</table>

*No entry was made if parameter not measured or not reported in convenient form. Representative value shown if range of values was reported.
Table 2. Characteristics of PWR-type specimens with Zircaloy-4 cladding tested in flowing water at an initial temperature of 265°C in the PBF test reactor.*

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Test Date</th>
<th>Maximum Burnup GWd/t</th>
<th>Oxide Thickness</th>
<th>Energy Deposit cal/g</th>
<th>Pulse Width ms</th>
<th>Peak Fuel Enthalpy cal/g</th>
<th>Cladding Strain %</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>RIA 802-1</td>
<td>11/22/78</td>
<td>5</td>
<td>5</td>
<td>240</td>
<td>16</td>
<td>185</td>
<td>3</td>
<td>No failure</td>
</tr>
<tr>
<td>RIA 802-2</td>
<td>11/22/78</td>
<td>5</td>
<td>5</td>
<td>240</td>
<td>16</td>
<td>185</td>
<td>6</td>
<td>No failure</td>
</tr>
<tr>
<td>RIA 802-3</td>
<td>11/22/78</td>
<td>4</td>
<td>5</td>
<td>240</td>
<td>16</td>
<td>185</td>
<td></td>
<td>PCMI failure at 140 cal/g</td>
</tr>
<tr>
<td>RIA 802-4</td>
<td>11/22/78</td>
<td>5</td>
<td>5</td>
<td>240</td>
<td>16</td>
<td>185</td>
<td>5</td>
<td>No failure</td>
</tr>
<tr>
<td>RIA ST-1</td>
<td>8/78</td>
<td>0</td>
<td>0</td>
<td>250</td>
<td>22</td>
<td>185</td>
<td></td>
<td>No failure</td>
</tr>
<tr>
<td>RIA ST-2</td>
<td>8/78</td>
<td>0</td>
<td>0</td>
<td>345</td>
<td>17</td>
<td>260</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RIA ST-3</td>
<td>8/78</td>
<td>0</td>
<td>0</td>
<td>300</td>
<td>20</td>
<td>225</td>
<td></td>
<td>No failure</td>
</tr>
<tr>
<td>RIA ST-4</td>
<td>8/78</td>
<td>0</td>
<td>0</td>
<td>695</td>
<td>30</td>
<td>350</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RIA 801-1</td>
<td>10/7/78</td>
<td>5</td>
<td>5</td>
<td>365</td>
<td>13</td>
<td>285</td>
<td></td>
<td>Rod fragmented and blocked flow channel during transient</td>
</tr>
<tr>
<td>RIA 801-2</td>
<td>10/7/78</td>
<td>5</td>
<td>5</td>
<td>365</td>
<td>13</td>
<td>285</td>
<td></td>
<td>Rod fragmented and blocked flow channel during transient</td>
</tr>
<tr>
<td>RIA 801-3</td>
<td>10/7/78</td>
<td>0</td>
<td>0</td>
<td>365</td>
<td>13</td>
<td>285</td>
<td></td>
<td>Rod fragmented and blocked flow channel after transient</td>
</tr>
<tr>
<td>RIA 801-5</td>
<td>10/7/78</td>
<td>0</td>
<td>0</td>
<td>365</td>
<td>13</td>
<td>285</td>
<td></td>
<td>Rod fragmented and blocked flow channel during transient</td>
</tr>
<tr>
<td>RIA 804-1</td>
<td>4/80</td>
<td>5</td>
<td>5</td>
<td>295</td>
<td>11</td>
<td>277</td>
<td></td>
<td>PCMI failure &lt;=255 cal/g</td>
</tr>
<tr>
<td>RIA 804-3</td>
<td>4/80</td>
<td>5</td>
<td>5</td>
<td>295</td>
<td>11</td>
<td>277</td>
<td></td>
<td>PCMI failure &lt;=255 cal/g</td>
</tr>
<tr>
<td>RIA 804-4</td>
<td>4/80</td>
<td>5</td>
<td>5</td>
<td>270</td>
<td>11</td>
<td>255</td>
<td></td>
<td>PCMI failure &lt;=255 cal/g</td>
</tr>
<tr>
<td>RIA 804-5</td>
<td>4/80</td>
<td>5</td>
<td>5</td>
<td>245</td>
<td>11</td>
<td>234</td>
<td></td>
<td>PCMI failure &lt;=255 cal/g</td>
</tr>
<tr>
<td>RIA 804-6</td>
<td>4/80</td>
<td>5</td>
<td>5</td>
<td>270</td>
<td>11</td>
<td>255</td>
<td></td>
<td>PCMI failure &lt;=255 cal/g</td>
</tr>
<tr>
<td>RIA 804-7</td>
<td>4/80</td>
<td>6</td>
<td>5</td>
<td>295</td>
<td>11</td>
<td>277</td>
<td></td>
<td>PCMI failure &lt;=255 cal/g</td>
</tr>
<tr>
<td>RIA 804-8</td>
<td>4/80</td>
<td>5</td>
<td>5</td>
<td>270</td>
<td>11</td>
<td>255</td>
<td></td>
<td>PCMI failure &lt;=255 cal/g</td>
</tr>
<tr>
<td>RIA 804-9</td>
<td>4/80</td>
<td>6</td>
<td>5</td>
<td>295</td>
<td>11</td>
<td>277</td>
<td></td>
<td>PCMI failure &lt;=255 cal/g</td>
</tr>
<tr>
<td>RIA 804-10</td>
<td>4/80</td>
<td>5</td>
<td>5</td>
<td>270</td>
<td>11</td>
<td>255</td>
<td></td>
<td>Cladding melted as the result of contact with rods 804-8 and 804-9</td>
</tr>
</tbody>
</table>

*No entry was made if parameter not measured or not reported in convenient form. Representative value shown if range of values was reported.
Table 3. Characteristics of PWR fuel specimens with Zircaloy-4 cladding (except as noted) tested in flowing sodium at an initial temperature of 280°C in the Cabri test reactor.*

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Test Date</th>
<th>Maximum Burnup GWd/t</th>
<th>Oxide Thickness</th>
<th>Energy Deposit cal/g fuel</th>
<th>Pulse Width ms</th>
<th>Peak Fuel Enthalpy cal/g</th>
<th>Cladding Strain %</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>REP-Na1</td>
<td>11/10/93</td>
<td>64</td>
<td>80</td>
<td>111</td>
<td>9.5</td>
<td>114</td>
<td></td>
<td>PCMI failure at 30 cal/g Significant fuel loss</td>
</tr>
<tr>
<td>REP-Na2</td>
<td>6/10/94</td>
<td>33</td>
<td>10</td>
<td>207</td>
<td>9.5</td>
<td>199</td>
<td>3.5</td>
<td>No failure</td>
</tr>
<tr>
<td>REP-Na3</td>
<td>10/6/94</td>
<td>54</td>
<td>45</td>
<td>122</td>
<td>9.5</td>
<td>124</td>
<td>2.2</td>
<td>No failure</td>
</tr>
<tr>
<td>REP-Na4</td>
<td>7/28/95</td>
<td>62</td>
<td>80</td>
<td>95</td>
<td>7.6</td>
<td>85</td>
<td>0.4</td>
<td>No failure</td>
</tr>
<tr>
<td>REP-Na5</td>
<td>5/5/95</td>
<td>64</td>
<td>25</td>
<td>104</td>
<td>8.8</td>
<td>108</td>
<td>1.1</td>
<td>No failure</td>
</tr>
<tr>
<td>REP-Na6</td>
<td>3/1/96</td>
<td>47</td>
<td>35</td>
<td>156</td>
<td>32</td>
<td>133</td>
<td>2.6</td>
<td>No failure, MOX**</td>
</tr>
<tr>
<td>REP-Na7</td>
<td>1/24/97</td>
<td>55</td>
<td>50</td>
<td>170</td>
<td>40</td>
<td>138</td>
<td></td>
<td>PCMI failure at 113 cal/g, MOX Fuel loss</td>
</tr>
<tr>
<td>REP-Na8</td>
<td>7/10/97</td>
<td>60</td>
<td>110</td>
<td>103</td>
<td>75</td>
<td>98</td>
<td></td>
<td>PCMI failure at 78 cal/g No fuel loss</td>
</tr>
<tr>
<td>REP-Na9</td>
<td>4/25/97</td>
<td>28</td>
<td>10</td>
<td>233</td>
<td>33</td>
<td>197</td>
<td>7.2</td>
<td>No failure, MOX</td>
</tr>
<tr>
<td>REP-Na10</td>
<td>7/30/98</td>
<td>63</td>
<td>80</td>
<td>108</td>
<td>31</td>
<td>98</td>
<td></td>
<td>PCMI failure at 81 cal/g No fuel loss</td>
</tr>
<tr>
<td>CIP0-1</td>
<td>11/29/02</td>
<td>75</td>
<td>80</td>
<td>98</td>
<td>32</td>
<td>90</td>
<td></td>
<td>No failure, Zirlo cladding</td>
</tr>
<tr>
<td>CIP0-2</td>
<td>11/8/02</td>
<td>77</td>
<td>20</td>
<td>89</td>
<td>28</td>
<td>81</td>
<td></td>
<td>No failure, M5 cladding</td>
</tr>
</tbody>
</table>

*No entry was made if parameter not measured or not reported in convenient form. Representative value shown if range of values was reported.
**Mixed uranium and plutonium dioxide (MOX).
Table 4. Characteristics of commercial PWR fuel specimens with Zircaloy-4 cladding (except as noted) tested in stagnant water at an initial temperature of 20°C in the NSRR test reactor.*

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Test Date</th>
<th>Maximum Burnup GWd/t</th>
<th>Oxide Thickness</th>
<th>Energy Deposit cal/g</th>
<th>Pulse Width ms</th>
<th>Peak Fuel Enthalpy cal/g</th>
<th>Cladding Strain %</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>MH-1</td>
<td>11/28/89</td>
<td>39</td>
<td>4</td>
<td>63</td>
<td>6.8</td>
<td>47</td>
<td>0.02</td>
<td>No failure</td>
</tr>
<tr>
<td>MH-2</td>
<td>3/8/90</td>
<td>39</td>
<td>4</td>
<td>72</td>
<td>5.5</td>
<td>54</td>
<td>0.05</td>
<td>No failure</td>
</tr>
<tr>
<td>MH-3</td>
<td>10/31/90</td>
<td>39</td>
<td>4</td>
<td>87</td>
<td>4.5</td>
<td>67</td>
<td>1.56</td>
<td>No failure</td>
</tr>
<tr>
<td>GK-1</td>
<td>3/12/91</td>
<td>42</td>
<td>10</td>
<td>121</td>
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<td>2.23</td>
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<tr>
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<td>15</td>
<td>139</td>
<td>4.4</td>
<td>108</td>
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</tr>
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<td>23</td>
<td>95</td>
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<td>80</td>
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<td>PCMI failure at 77 cal/g</td>
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</tr>
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<td>109</td>
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<td>45</td>
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<td>88</td>
<td>2.23</td>
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<td>126</td>
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<td>35</td>
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<td>107</td>
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<td>PCMI failure at 60 cal/g</td>
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<td></td>
<td>Small fuel loss</td>
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<td>TK-3</td>
<td>10/8/97</td>
<td>50</td>
<td>10</td>
<td>126</td>
<td>4.4</td>
<td>99</td>
<td>5.6</td>
<td>No failure</td>
</tr>
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<td>2/4/98</td>
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<td>25</td>
<td>125</td>
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<td>98</td>
<td>4</td>
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<td>10/1/98</td>
<td>48</td>
<td>30</td>
<td>130</td>
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<td>101</td>
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<tr>
<td>TK-6</td>
<td>10/7/98</td>
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<td>15</td>
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<td>122</td>
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<td>95</td>
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<td>No report on fuel loss</td>
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<td>65</td>
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<td>126</td>
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<td>99</td>
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</tr>
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<td>&lt;10</td>
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<td>27</td>
<td>138</td>
<td>5.6</td>
<td>104</td>
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</tr>
<tr>
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<td>58</td>
<td>28</td>
<td>201</td>
<td>4.4</td>
<td>157</td>
<td></td>
<td>PCMI failure at 120 cal/g, Zirlo cladding</td>
</tr>
</tbody>
</table>

*No entry was made if parameter not measured or not reported in convenient form. Representative value shown if range of values was reported.
Table 5. Characteristics of PWR-type fuel specimens with Zircaloy-4 cladding that were irradiated in material test reactors and tested in stagnant water at an initial temperature of 20°C in the NSRR test reactor.*

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Test Date</th>
<th>Maximum Burnup GWd/t</th>
<th>Oxide Thickness</th>
<th>Energy Deposit cal/g</th>
<th>Pulse Width ms</th>
<th>Peak Fuel Enthalpy cal/g</th>
<th>Cladding Strain %</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>JM-1</td>
<td>7/20/89</td>
<td>22</td>
<td>&lt;2</td>
<td>130</td>
<td>9.3</td>
<td>92</td>
<td>0.1</td>
<td>No failure</td>
</tr>
<tr>
<td>JM-2</td>
<td>1/11/90</td>
<td>27</td>
<td>&lt;2</td>
<td>120</td>
<td>9.4</td>
<td>84</td>
<td>0</td>
<td>No failure</td>
</tr>
<tr>
<td>JM-3</td>
<td>9/6/90</td>
<td>20</td>
<td>&lt;2</td>
<td>184</td>
<td>8.5</td>
<td>132</td>
<td>0.4</td>
<td>No failure</td>
</tr>
<tr>
<td>JM-4</td>
<td>11/6/90</td>
<td>21</td>
<td>&lt;2</td>
<td>235</td>
<td>5.9</td>
<td>177</td>
<td>Failure enthalpy not reported No fuel loss</td>
<td></td>
</tr>
<tr>
<td>JM-5</td>
<td>3/5/91</td>
<td>26</td>
<td>&lt;2</td>
<td>220</td>
<td>6.4</td>
<td>167</td>
<td>Failure enthalpy not reported No fuel loss</td>
<td></td>
</tr>
<tr>
<td>JM-6</td>
<td>2/13/92</td>
<td>15</td>
<td>&lt;2</td>
<td>212</td>
<td>7.1</td>
<td>156</td>
<td>1</td>
<td>No failure</td>
</tr>
<tr>
<td>JM-7</td>
<td>9/9/92</td>
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<td>&lt;2</td>
<td>201</td>
<td>7.8</td>
<td>146</td>
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<td>No failure</td>
</tr>
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<td>9/17/92</td>
<td>20</td>
<td>&lt;2</td>
<td>218</td>
<td>7.2</td>
<td>160</td>
<td>1.6</td>
<td>No failure</td>
</tr>
<tr>
<td>JM-9</td>
<td>3/30/93</td>
<td>25</td>
<td>&lt;2</td>
<td>210</td>
<td>6.8</td>
<td>160</td>
<td>0.7</td>
<td>No failure</td>
</tr>
<tr>
<td>JM-10</td>
<td>9/27/93</td>
<td>21</td>
<td>&lt;2</td>
<td>270</td>
<td>5.6</td>
<td>200</td>
<td>7.7</td>
<td>No failure</td>
</tr>
<tr>
<td>JM-11</td>
<td>12/1/93</td>
<td>31</td>
<td>&lt;2</td>
<td>210</td>
<td>6.3</td>
<td>160</td>
<td>0.9</td>
<td>No failure</td>
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<td>11/25/93</td>
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<td>240</td>
<td>5.3</td>
<td>180</td>
<td>Failure enthalpy not reported No fuel loss</td>
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</tr>
<tr>
<td>JM-13</td>
<td>1/18/95</td>
<td>38</td>
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<td>200</td>
<td>6.3</td>
<td>150</td>
<td>0.7</td>
<td>No failure</td>
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<tr>
<td>JM-14</td>
<td>3/7/95</td>
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<td>160</td>
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<td>150</td>
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<td>No failure</td>
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<tr>
<td>JM-16</td>
<td>10/17/95</td>
<td>38</td>
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<td>140</td>
<td>3.2</td>
<td>No failure</td>
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<td>11/17/92</td>
<td>22</td>
<td>&lt;2</td>
<td>190</td>
<td>7.1</td>
<td>150</td>
<td>Failure enthalpy not reported No fuel loss</td>
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<tr>
<td>JMH-1</td>
<td>3/23/93</td>
<td>22</td>
<td>&lt;2</td>
<td>200</td>
<td>8.3</td>
<td>150</td>
<td>1.7</td>
<td>No failure</td>
</tr>
<tr>
<td>JMH-2</td>
<td>9/30/93</td>
<td>22</td>
<td>&lt;2</td>
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<td>180</td>
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<td>No failure</td>
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<td>3/13/95</td>
<td>30</td>
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<td>270</td>
<td>6.2</td>
<td>220</td>
<td>PCMI failure at 205 cal/g Fuel loss</td>
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<td>&lt;2</td>
<td>190</td>
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<td>170</td>
<td>3.6</td>
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<td>220</td>
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<td>80</td>
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<td>3/14/97</td>
<td>20</td>
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<td>155</td>
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<td>120</td>
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<td>4.4</td>
<td>85</td>
<td>No failure, MOX</td>
<td></td>
</tr>
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</table>

*No entry was made if parameter not measured or not reported in convenient form. Representative value shown if range of values was reported.
Table 6. Characteristics of BWR fuel specimens with Zircaloy-2 cladding tested in stagnant water at an initial temperature of 20°C in the NSRR test reactor.*

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Test Date</th>
<th>Maximum Burnup (GWd/t)</th>
<th>Oxide Thickness :</th>
<th>Energy Deposit (cal/g)</th>
<th>Pulse Width (ms)</th>
<th>Peak Fuel Enthalpy (cal/g)</th>
<th>Cladding Strain (%)</th>
<th>Comments</th>
</tr>
</thead>
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<tr>
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<td>10/24/89</td>
<td>26</td>
<td>6</td>
<td>70</td>
<td>6.0</td>
<td>55</td>
<td>0</td>
<td>No failure</td>
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<td>2/7/90</td>
<td>26</td>
<td>6</td>
<td>82</td>
<td>5.3</td>
<td>66</td>
<td>0</td>
<td>No failure</td>
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<tr>
<td>TS-3</td>
<td>9/12/90</td>
<td>26</td>
<td>6</td>
<td>109</td>
<td>4.8</td>
<td>88</td>
<td>0</td>
<td>No failure</td>
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<tr>
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<td>1/17/91</td>
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<td>6</td>
<td>110</td>
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<td>89</td>
<td>0.47</td>
<td>No failure</td>
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<td>6</td>
<td>117</td>
<td>4.4</td>
<td>98</td>
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<td>11/21/96</td>
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<td>16</td>
<td>167</td>
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<td>130</td>
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<td>45</td>
<td>18</td>
<td>95</td>
<td>6.6</td>
<td>70</td>
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<td>No failure</td>
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<tr>
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<td>22</td>
<td>100</td>
<td>7.3</td>
<td>70</td>
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<td>No failure</td>
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<td>FK-6</td>
<td>3/7/00</td>
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<td>20 - 30</td>
<td>168</td>
<td>4.4</td>
<td>131</td>
<td>PCMI failure at 70 cal/g Fuel loss</td>
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<tr>
<td>FK-7</td>
<td>3/14/00</td>
<td>61</td>
<td>20 - 30</td>
<td>166</td>
<td>4.4</td>
<td>129</td>
<td>PCMI failure at 62 cal/g No report on fuel loss</td>
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</tr>
<tr>
<td>FK-8</td>
<td>10/5/00</td>
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<td>20 - 30</td>
<td>90</td>
<td>7.3</td>
<td>65</td>
<td>0.02</td>
<td>No failure</td>
</tr>
<tr>
<td>FK-9</td>
<td>11/1/00</td>
<td>61</td>
<td>20 - 30</td>
<td>119</td>
<td>5.7</td>
<td>90</td>
<td>PCMI failure at 86 cal/g Fuel loss</td>
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<td>FK-10</td>
<td>10/12/01</td>
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<td>20 - 30</td>
<td>135</td>
<td>5.1</td>
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<td>PCMI failure at 80 cal/g No report on fuel loss</td>
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<td>FK-12</td>
<td>12/13/02</td>
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<td>20 - 30</td>
<td>118</td>
<td>5.8</td>
<td>89</td>
<td>PCMI failure at 72 cal/g No report on fuel loss</td>
<td></td>
</tr>
</tbody>
</table>

*No entry was made if parameter not measured or not reported in convenient form. Representative value shown if range of values was reported.
Table 7. Characteristics of VVER fuel specimens with E110 cladding tested in stagnant water at an initial temperature of 20°C in the IGR test reactor.*

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Test Date</th>
<th>Maximum Burnup GWD/t</th>
<th>Oxide Thickness:</th>
<th>Energy Deposit cal/g</th>
<th>Pulse Width ms</th>
<th>Peak Fuel Enthalpy cal/g</th>
<th>Cladding Strain %</th>
<th>Comments</th>
</tr>
</thead>
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<tr>
<td>H1T</td>
<td>1990-92</td>
<td>49</td>
<td>5</td>
<td>253</td>
<td>800</td>
<td>151</td>
<td>1.4</td>
<td>No failure</td>
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<td>H2T</td>
<td>1990-92</td>
<td>48</td>
<td>5</td>
<td>333</td>
<td>760</td>
<td>213</td>
<td></td>
<td>Swelling and rupture; failure time not measured No observation for fuel loss</td>
</tr>
<tr>
<td>H3T</td>
<td>1990-92</td>
<td>49</td>
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*No entry was made if parameter not measured or not reported in convenient form. Representative value shown if range of values was reported.
Table 8. Characteristics of VVER fuel specimens with E110 cladding tested in stagnant water at an initial temperature of 20°C in the BIGR test reactor.*

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Test Date</th>
<th>Maximum Burnup GWd/t</th>
<th>Oxide Thickness:</th>
<th>Energy Deposit cal/g</th>
<th>Pulse Widths ms</th>
<th>Peak Fuel Enthalpy cal/g</th>
<th>Cladding Strain %</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>RT-1</td>
<td>1997-00</td>
<td>48</td>
<td>5</td>
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<td>142</td>
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<td>48</td>
<td>5</td>
<td>143</td>
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<td>146</td>
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<td>201</td>
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<td>155</td>
<td>5</td>
<td>No failure</td>
</tr>
</tbody>
</table>

*No entry was made if parameter not measured or not reported in convenient form. Representative value shown if range of values was reported.
Table 9. Summary of results from scaling analysis.

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Calculation for Test Conditions</th>
<th>Calculation for PWR Conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Initial Temp (°C)</td>
<td>Pulse Width (ms)</td>
</tr>
<tr>
<td>REP-Na10</td>
<td>280</td>
<td>31</td>
</tr>
<tr>
<td>REP-Na8</td>
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<td>REP-Na7</td>
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<td>40</td>
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<tr>
<td>HBO-1</td>
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<tr>
<td>TK-2</td>
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<td>4.4</td>
</tr>
</tbody>
</table>
Figure 3

Pulse Width (ms) vs. Maximum Fuel Enthalpy Change (cal/g)

- △ EOC
- ▲ BOC
Figure 4
Figure 5
Figure 6
Figure 8
Figure 9
Figure 10
Figure 11
Figure 14

Max. Fuel Enthalpy Change (cal/g) vs. Ejected Control Rod Worth ($) for different Beta levels:
- EOC 120% Beta
- EOC 110% Beta
- EOC 100% Beta
- EOC 90% Beta
- EOC 80% Beta
- EOC 70% Beta
- BOC 100% Beta

The graph shows a significant increase in max. fuel enthalpy change with an increase in ejected control rod worth for all Beta levels.