Draft Interim Review of PRM-50-93/95 Issues Related to Loss-of-Coolant Accident Simulations in the National Research Universal (NRU) Reactor

Disclaimer:

Public availability of this draft interim review is intended to inform stakeholders of the current status of the NRC review of the issues raised in PRM-50-93/95. This draft interim review is subject to further revisions during resolution of PRM-50-93/95. The NRC is not soliciting public comments on these interim conclusions, and will not provide a formal response to any comments received. The NRC findings on PRM-50-93/95 issues will not be final until the NRC publishes a notice of final action on this petition for rulemaking in the *Federal Register*.

1.0 Issues Raised in the Petitions and Associated Comments

A petition for rulemaking was docketed as PRM-50-93 on November 17, 2009 (Leyse, 2009). The petitioner is requesting revisions to section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR) "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" and to 10 CFR Part 50, Appendix K "ECCS Evaluation Models," as well as associated regulatory guidance. The petitioner, Mark Edward Leyse, discussed the Zircaloy assembly tests performed at the National Research Universal (NRU) Reactor as part of the basis for the assertion in the petition that 1) low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases and 2) the zirconium-water reaction equations (Baker-Just and Cathcart-Pawel) are both non-conservative for calculating the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would occur in the event of a LOCA.

Petition Arguments

The petitioner raises several issues related to the NRU thermal-hydraulic tests. The first of which is related to the reflood rate. Specifically, the petition states:

> "it can be extrapolated from experimental data that, in the event a LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F."

> "The NRU Thermal-Hydraulic Experiment 1 ("TH-1") tests illustrate that low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases."

The petitioner then goes on to present temperature increases and peak cladding temperatures for several NRU tests with different reflood rates. Petitioner then states that if these results are extrapolated to some hypothetical set of conditions, that the PCT would exceed the 2,200°F limit.

The second issue related to the NRU tests is use of the Baker-Just and Cathcart-Pawel metal-water reaction equations. The petitioner makes a comparison between the data from NRU TH-1 test 128 and a sample code prediction (using the Baker-Just correlation) from Westinghouse's "PWR FLECHT Final Report." Based on results of this comparison, petitioner states:

> "it is evident that analyses using the Baker-Just correlation under-predict the amount of heat generated by Zircaloy oxidation in TH-1 test no. 128."

> "So with similar parameters (but with a lower fuel rod power) TH-1 test no. 128 had an overall PCT increase that was more than 100°F greater than the overall PCT increase predicted in the UO2 Zircaloy fuel assembly example discussed in "PWR FLECHT Final Report." This indicates that analyses using the Baker-Just correlation under-predict the amount of heat that Zircaloy oxidation generated in TH- 1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions."

The petitioner also discusses TH-1 test 130. The petitioner states:

> "the reactor shutdown when the PCT was approximately 1850°F; and after the reactor shutdown, cladding temperatures kept increasing because of the heat generated from the Zircaloy-steam reaction (of course, there would have also been a small amount of actual decay heat) and the peak measured cladding temperature was 2040°F. So the peak cladding temperature increased by 190°F after the reactor shutdown, because of the heat generated from the Zircaloy-steam reaction."

> "It is highly unlikely that analyses using the Baker-Just and Cathcart-Pawel correlations would predict a peak cladding temperature increase of 190°F in TH-1 test no. 130, after the reactor shutdown."

> "So data from thermal hydraulic experiments indicates that the Baker-Just and Cathcart-Pawel correlations are not adequate for use in ECCS evaluation calculations that calculate the metal-water reaction rates that would occur in the heat transfer conditions of loss-of-coolant accidents."

2.0 Staff Response to Assertions

2.1 Reflood Rate

The petitioner states that the TH-1 tests illustrate that low reflood rates do not prevent Zircaloy cladding temperatures from having substantial increases. There are numerous parameters that are known to have an important effect on reflood hydraulics in a light water reactor. No single parameter completely controls the peak cladding temperature for a particular transient. Basing a conclusion on any single parameter can be misleading. Part of the basis for the petitioners request for a limit on reflood rate, is the significant temperature increases observed in the NRU reflood tests. Starting from initial cladding temperatures less than 1,000°F, several NRU tests produced temperature increases of over 1,000°F. The petition cites several NRU tests as examples (tests no. 127 and 130). The petition implies that similar, or larger, temperature increases would occur if the initial cladding temperatures had been 1,200°F or more. This is not correct, however. Thermal radiation becomes more important in transferring heat away from hot spots, and as rod temperatures increase the temperature difference between the cladding and the coolant increases. Figure 1 below shows the temperature at the start of reflood –vs– the temperature rise for all the NRU TH-1 tests. These include various reflood rates and delay times. As seen in the figure, as the cladding temperature at the start of reflood increases, the overall temperature rise decreases. The red line on the figure shows the margin to a PCT of 2,200°F. Thus, contrary to the claim made in the petition, "extrapolation" of data does not show "with high probability" that peak cladding temperatures will exceed 2,200°F.

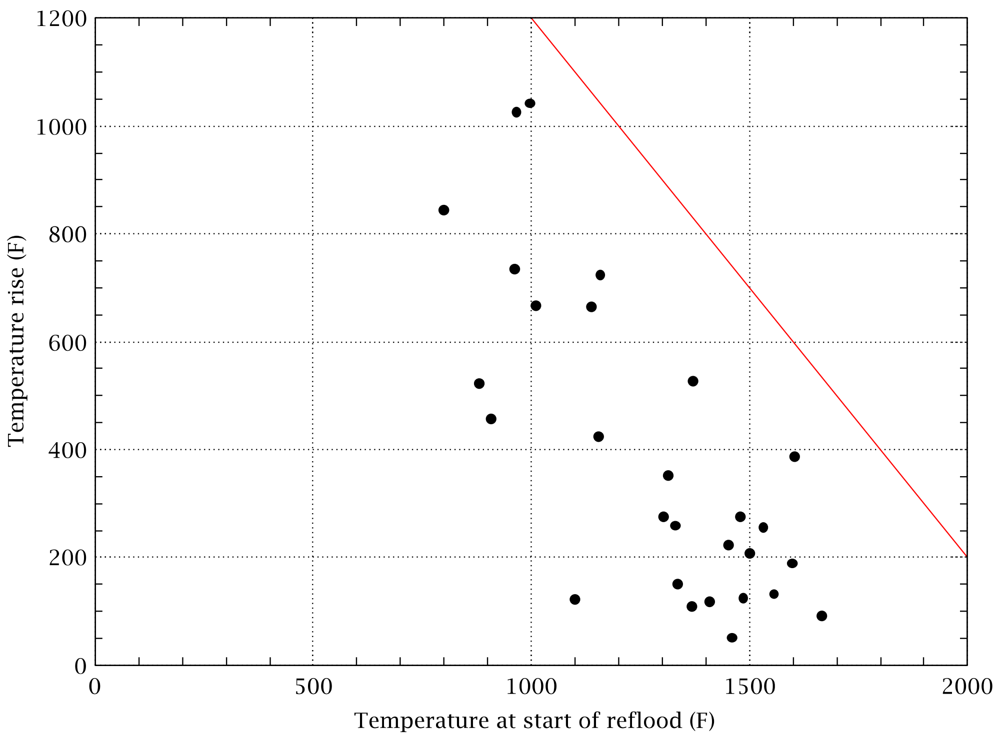

Figure 1. Temperature at start of reflood –vs- temperature rise for all NRU TH-1 tests

2.2 Metal-water reaction

<u>NRU Test no 128.</u>

The petitioner argues that the Baker-Just metal-water reaction correlation is non-conservative based on comparison of the NRU TH-1 test no. 128 against some PWR FLECHT code calculations. The petitioner specifically states:

> "So with similar parameters (but with a lower fuel rod power) TH-1 test no. 128 had an overall PCT increase that was more than 100°F greater than the overall PCT increase predicted in the UO2 Zircaloy fuel assembly example discussed in "PWR FLECHT Final Report." This indicates that analyses using the Baker-Just correlation under-predict the amount of heat that Zircaloy oxidation generated in TH-1 test no. 128, a thermal hydraulic experiment simulating LOCA conditions."

The calculations that the petitioner is comparing TH-1 test no. 128 against are described in Section 4 of WCAP-7665. The code calculations were performed to show the difference between electrically heated rods with either stainless steel or Zr cladding and UO_2 heated rods with Zr cladding under a hypothetical LOCA with two different reflood rates. The calculations were done with an unnamed "conduction code" which includes the Baker-Just parabolic rate equation. It is the staffs' opinion that this type of indirect comparison (between code results for one facility used for a specific purpose and experimental data from a specific test in a different test facility) should not form the basis for any decisions when direct comparisons are available and can be investigated. However, the results of the PWR FLECHT code runs are still useful and comparisons can be made between PWR FLECHT experimental data and NRU experimental data (TH-1 test no 128).

While the petitioner compared NRU TH-1 test no. 128 against some PWR FLECHT code calculations, the staff compared test no. 128 against PWR FLECHT run no. 4129 (Table 1 below). As seen in the table, the parameters and results from the two tests are similar. Both tests started with an initial temperature around 1,600°F and reflood rate around 2 in/s. In the case of the PWR FLECHT test with the stainless steel rods, the temperature at the hot rod mid-plane elevation increased 372°F while the TH-1 test with Zr rods increased 387°F. The small difference in temperature rise between the two cases (15°F) shows that the zirconium metal-water reaction is not substantial at these temperatures (just below 2,000°F).

Table 1. Comparison of selected TH-1 and PWR FLECHT tests

Parameter	TH-1 test no. 128	PWR FLECHT run no. 4129
Cladding material	Zr	Stainless steel
Flooding rate	2 in/s	1.9 in/s
Initial temperature[1]	1,604°F	1,596°F
Temperature rise	387°F	372°F
Peak clad temperature[1]	1,991°F	1,968°F

[1] The FLECHT temperature is not the peak clad temperature, rather it is the mid-plane temperature

The PWR FLECHT code calculations referred to by the petitioner were not performed to simulate a specific test, rather they were performed to show the difference between electrically heated rods with either stainless steel or Zr cladding and UO_2 heated rods with Zr cladding under a hypothetical LOCA. However, the conditions are very similar to PWR FLECHT run no. 4225 and comparisons are shown in Table 2. In the case of the stainless steel fuel rods, there is no metal-water reaction simulated and the code results are with a few °F of the experiment. When UO_2 heated rods with Zr cladding are used in the code, the temperature rise was 27°F larger than the stainless steel code run. This difference is mainly due to the metal-water reaction (using the Baker-Just metal-water correlation). This difference is larger than the difference of 15°F seen when comparing TH-1 test no. 128 and PWR FLECHT run no. 4129.

Based on comparisons of NRU TH-1 test no 128, PWR FLECHT run nos. 4129 and 4225 along with the PWR FLECHT code calculations referenced by the petitioner, there is no evidence that the Baker-Just (or Cathcart-Pawel) metal-water reaction correlation is non-conservative, or that the metal-water reaction is significant at temperatures < 2,000°F as asserted in the petition.

Table 2. Comparison of PWR FLECHT test and code results

Parameter	PWR FLECHT run no. 4225[1]	PWR FLECHT code calculation[2]	PWR FLECHT code calculation[2]
Cladding material	Stainless steel	Stainless steel	Zr
Flooding rate	1.9 in/s	2 in/s	2 in/s
Pressure	59 psia	60 psia	60 psia
Peak power	1.24 kW/ft	1.24 kW/ft	1.24 kW/ft
Initial temperature	1,596°F	1,600°F	1,600°F
Temperature rise	247°F	244°F	271°F
Peak clad temperature	1,843°F	1,844°F	1,871°F

[1] Data from Table 3-1 of WCAP-7665
[2] Data from Figure 4-1 of WCAP-7665

NRU Test no 130.

The petitioner argues that the experimental data from multi-rod thermal hydraulic experiments (specifically TH1 test no. 130) demonstrates that the Zircaloy-steam reaction is substantial below 1,900°F. The petitioner states:

> "In TH-1 test no. 130, the reactor tripped (shutdown) when the PCT was approximately 1,850°F; and after the reactor shutdown, cladding temperatures kept increasing because of the heat generated from the metal-water reaction (of course, there would have also been a small amount of actual decay heat) and the peak measured cladding temperature was 2,040°F. So because of the heat generated from the metal-water reaction, the peak cladding temperature increased by 190°F, after the reactor shutdown."

The values quoted by the petitioner were obtained from NUREG/CR-1882 (Table 1 on page 13). In NUREG/CR-1882, it states in a footnote to Table 1 for test no. 130, "Reactor tripped at ~1,850°F." The petitioners claim is based on this temperature being the peak cladding temperature, however, this is not correct. Section 9.7 of NUREG/CR-1208 describes the reactor trip conditions and requirements as well as the neutron power decay. There are several transient phase trips, including:

- Hanger tube high temperature trip.
- Outlet piping high temperature trip.
- Low reflood flow standby trip.
- High fuel cladding temperature trip.
- Manual trip.

The description for the fuel cladding temperature trip states that the temperature signal used is an average of the guard fuel rods (located on the peripheral region of the assembly) and not the test fuel rods (located in the center region of the assembly). This average of all the guard fuel rod thermocouples (at a given axial level) is then averaged over a ten second time period, resulting in a lag of the actual temperature response. Given the fact that the peak temperature will occur in the test fuel rods and the fact that the temperature signal used in the trip logic is a lagged average of the guard fuel rods, the peak cladding temperature will be larger than the trip setpoint of 1,850°F at the time of trip. In addition, there is an approximately 0.75 second delay after the initiation of trip before the power begins to decline as seen in Figure 9.3 of NUREG/CR-1208. Therefore, since the temperatures are all increasing, by definition the peak cladding temperature must be larger than the 1,850°F reactor trip setpoint when the reactor power begins to decline.

To investigate this further, applicable portions of several temperature plots for TH-1 test no. 130 (shown in Figures 1.2.3.30, 1.2.4.30 and 3.2.2.30 of the Appendix to NUREG/CR-1882) were digitized and recreated and shown in Figure 2 below. The highest temperature shown in the plots for test no. 130 is 1,968°F, which is lower than the 2,040°F given in Table 1 of NUREG/CR-1882, so the peak cladding temperature location is not shown in the plots and it cannot be determined exactly how much the peak cladding temperature increased after reactor power was stopped. While it is not known at exactly what time the reactor was tripped, it would be expected that there would be some change in the rate of temperature increase when the reactor trips and power begins to decline. Looking at the slope of the temperature curves (dT/dt) shows no noticeable change in any dT/dt prior to 100 s. While the temperature may continue to increase at some locations (due to the metal-water reaction), it is extremely unlikely that it would continue to increase at the same rate at all locations shown in the temperature plots, particularly those below 1,800°F. Assuming the reactor was tripped and power began decreasing at 102 s (the first observed change in dT/dt), then thermocouple TC153B1R4 would be at 1,940°F and reach a maximum of 1,968°F, a difference of 28°F. While this is not the location of the peak cladding temperature, this is the thermocouple with the highest temperature plotted in NUREG/CR-1882 and is representative of the thermal transient. This is a significantly smaller temperature increase than the 190°F quoted by the petitioner. Given that the reactor did not trip at a PCT of 1,850°F, there is no basis for the petitioners' claim that a temperature increase of 190°F was due to the metal-water reaction after the reactor shutdown.

The petitioner also states:

> "It is clear that, in TH-1 test no. 130, if the reactor had not shutdown when the PCT was approximately 1850°F, that the overall PCT would have been greater than 2040°F. In fact, it is highly probable that the multi-rod bundle in the TH-1 test no. 130, would have incurred autocatalytic oxidation if the reactor had not shutdown when the PCT was approximately 1850°F."

While the peak cladding temperature would have gone higher than 2,040°F if the reactor was not tripped, there is no justification or evidence provided by the petitioner to substantiate the claim that autocatalytic oxidation would have occurred.

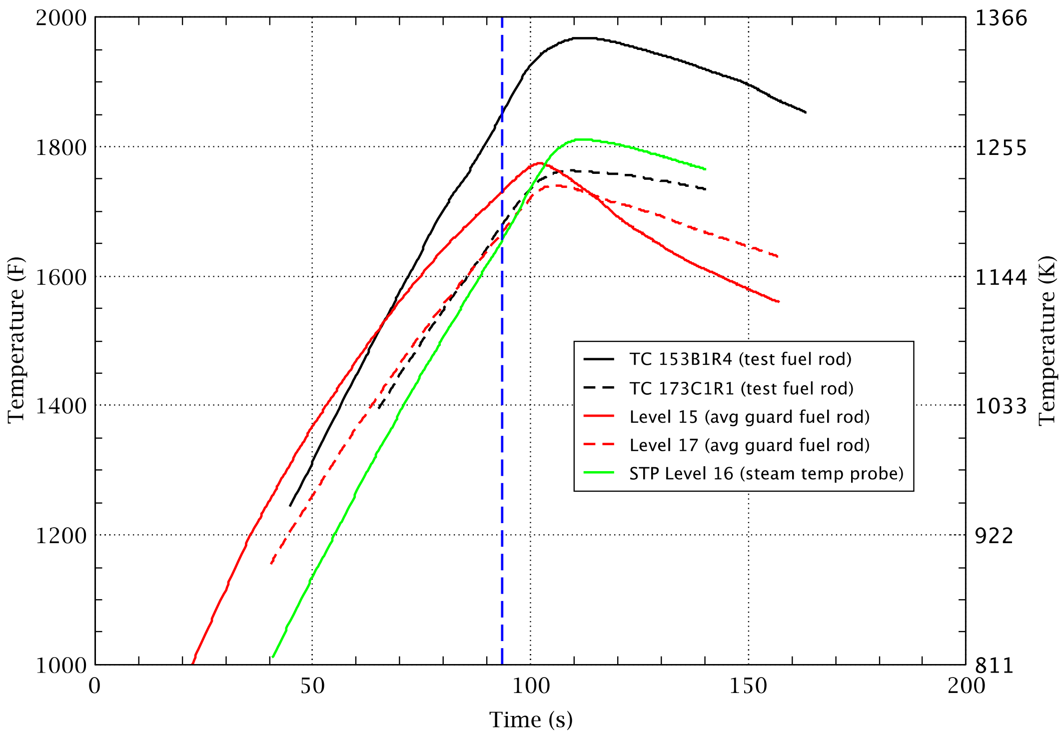

Figure 2. Temperatures for NRU TH-1 Test 130 from NUREG/CR-1882

3.0 Summary and Conclusions

This draft interim evaluation examined the NRU thermal hydraulic tests for issues related to reflood and the metal-water reaction. In the case of reflood, the petitioner claims through extrapolation that a high temperature at the start of reflood along with a low reflood rate will result in a large temperature rise. This extrapolation was shown to be invalid as the temperature increase declines as the temperature at the start of reflood increases due to effects not considered by the petitioner.

For the metal-water reaction, the petitioner argues that the Baker-Just correlation is non-conservative based on comparison of the NRU TH-1 test no. 128 against some PWR FLECHT code calculations. The petitioner was making an indirect comparison by comparing code results from one test facility (used for a specific purpose) to experimental results from a specific test in another facility. Staff compared NRU TH-1 test no. 128 against PWR FLECHT test 4129 and found only a small difference in PCT between the test with stainless steel rods and with Zr rods, demonstrating that the metal-water reaction is not significant at temperatures below 2,000°F. The PWR FLECHT code calculations were then compared against PWR FLECHT test 4225 and showed excellent agreement for stainless steel rods. When Zr rods were simulated in the code, the PCT increased by 27°F (mainly due to the metal-water reaction). A comparison between TH-1 test no. 128 (with Zr cladding) and PWR FLECHT run no. 4129 (with stainless steel cladding) showed a 15°F difference in temperature rise with the Zr rods, demonstrating that the metal-water reaction is not significant at these temperatures. Based on these comparisons, there is no evidence that the Baker-Just metal-water reaction correlation is non-conservative.

The petitioner argues that the experimental data from TH1 test no. 130 demonstrates that the Zircaloy-steam reaction is substantial below 1,900°F. The petitioner claims that the peak cladding temperature increased 190°F after the reactor was tripped. TH1 test no. 130 was examined and it was determined that the reactor was not tripped at a peak cladding temperature of 1,850°F as the petitioner claims, rather the 1,850°F trip setpoint is based on a time lagged average of the guard fuel rods. While the exact peak cladding temperature at the time reactor power begins to decline is unknown, it is by necessity higher than the 1,850°F trip setpoint, therefore, there is no basis for the statement that the peak cladding temperature increased 190°F after the reactor was tripped. There is also no justification or evidence provided by the petitioner to substantiate the claim that autocatalytic oxidation would have occurred had the reactor not been tripped.

4.0 References

Leyse, M. E., Petition for Rulemaking (Docketed as PRM 50-93), November 17, 2009, ADAMS Accession No. ML093290250

C. L. Mohr, et al., Pacific Northwest Laboratory, "Safety Analysis Report: Loss-of-Coolant Accident Simulations in the National Research Universal Reactor," NUREG/CR-1208, PNL-3093, 1981.

C. L. Mohr, et al., Pacific Northwest Laboratory, "Prototypic Thermal-Hydraulic Experiment in NRU to Simulate Loss-of-Coolant Accidents," NUREG/CR-1882, PNL-3681, 1981.

F. F. Cadek, et al., "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," WCAP-7665, 1971.

M. Leyse, Comments on PRM-50-93 and PRM-50-95; NRC-2009-0554, July 30, 2011, ADAMS accession number ML11213A211.

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