From: Mark Leyse [markleyse@gmail.com]
Sent: Tuesday, November 23, 2010 11:38 PM
To: Rulemaking Comments; PDR Resource
Cc: Dave Lochbaum; necnp@necnp.org; Raymond Shadis; Powers, Dana A
Subject: NRC-2009-0554 (Second)
Attachments: Comments November 2010 II.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is my second response, dated November 24, 2010, to the NRC's notice of solicitation of public comments on
PRM-50-93 and PRM-50-95, NRC-2009-0554, published in the Federal Register on October 27, 2010.

Sincerely,

Mark Leyse

DOCKETED
USNRC

November 24, 2010 (9:15am)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

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November 24, 2010

Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemakings and Adjudications Staff

COMMENTS ON PRM-50-93 AND PRM-50-95; NRC-2009-0554

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Annette L. Vietti-Cook
Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemakings and Adjudications Staff

COMMENTS ON PRM-50-93 AND PRM-50-95; NRC-2009-0554

I. Statement of Petitioner's Interest

On November 17, 2009, Mark Edward Leyse, Petitioner (in these comments "Petitioner" means Petitioner for PRM-50-93 and sole author of PRM-50-95), submitted a petition for rulemaking, PRM-50-93 (ADAMS Accession No. ML093290250). PRM-50-93 requests that the Nuclear Regulatory Commission ("NRC") make new regulations: 1) to require that the calculated maximum fuel element cladding temperature not exceed a limit based on data from multi-rod (assembly) severe fuel damage experiments;[1] and 2) to stipulate minimum allowable core reflood rates, in the event of a loss-of-coolant accident ("LOCA").[2, 3]

Additionally, PRM-50-93 requests that the NRC revise Appendix K to Part 50—ECCS Evaluation Models I(A)(5), *Required and Acceptable Features of the Evaluation Models*, *Sources of Heat during the LOCA*, *Metal-Water Reaction Rate*, to require that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-

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[1] Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the current 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F is non-conservative.

[2] It can be extrapolated from experimental data that, in the event a large break ("LB") LOCA, a constant core reflood rate of approximately one inch per second or lower (1 in./sec. or lower) would not, with high probability, prevent Zircaloy fuel cladding, that at the onset of reflood had cladding temperatures of approximately 1200°F or greater and an average fuel rod power of approximately 0.37 kW/ft or greater, from exceeding the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F. In the event of a LB LOCA, there would be variable reflood rates throughout the core; however, at times, local reflood rates could be approximately one inch per second or lower.

[3] It is noteworthy that in 1975, Fred C. Finlayson stated, "[r]ecommendations are made for improvements in criteria conservatism, especially in the establishment of minimum reflood heat transfer rates (or alternatively, reflooding rates);" see Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, Abstract, p. iii.

water reaction considered in ECCS evaluation calculations be based on data from multi-rod (assembly) severe fuel damage experiments.[4] These same requirements also need to apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.[5]

On June 7, 2010, Petitioner, submitted a 10 C.F.R. § 2.206 petition on behalf of New England Coalition ("NEC"), requesting that NRC order the licensee of Vermont Yankee Nuclear Power Station ("VYNPS") to lower the licensing basis peak cladding temperature ("LBPCT") of VYNPS in order to provide a necessary margin of safety—to help prevent a partial or complete meltdown—in the event of a loss-of-coolant accident ("LOCA").

On October 27, 2010, NRC published in the Federal Register a notice stating that it had determined that the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner submitted on behalf of NEC, meets the threshold sufficiency requirements for a petition for rulemaking under 10 C.F.R. § 2.802: NRC docketed the 10 C.F.R. § 2.206 petition as a petition for rulemaking, PRM-50-95 (ADAMS Accession No. ML101610121).[6]

When Petitioner wrote the 10 C.F.R. § 2.206 petition, dated June 7, 2010, Petitioner did not foresee that NRC would docket it as PRM-50-95. PRM-50-95 was written and framed as a 10 C.F.R. § 2.206 petition, not as a 10 C.F.R. § 2.802 petition; however, it is laudable that NRC is reviewing the issues Petitioner raised in PRM-50-95.

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[4] Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the temperature at which an autocatalytic (runaway) oxidation reaction of Zircaloy would commence in the event of a LOCA. This, in turn, indicates that the Baker-Just and Cathcart-Pawel correlations are both non-conservative for use in analyses that would predict the metal-water reaction rates that would occur in the event of a LOCA.

[5] Best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations are described in NRC Regulatory Guide 1.157.

[6] Federal Register, Vol. 75, No. 207, Notice of consolidation of petitions for rulemaking and re-opening of comment period, October 27, 2010, pp. 66007-66008.

II. Supplementary Information to PRM-50-93 and PRM-50-95

A. Presentation of Robert Leyse and Mark Leyse in Advisory Committee on Reactor Safeguards, Thermal Hydraulic Phenomena Subcommittee Meeting, October 18, 2010

A presentation that Robert Leyse and Mark Leyse gave in Advisory Committee on Reactor Safeguards ("ACRS"), Thermal Hydraulic Phenomena Subcommittee Meeting, on October 18, 2010, helps summarize some of the safety issues raised in PRM-50-93 and PRM-50-95.

The ACRS presentation is quoted below (with changes to some of the punctuation recorded in the transcript and changes to a few words that were improperly recorded):

Mark Leyse: First, I want to thank ACRS for the 10-minute time slot. Ten minutes is not a lot of time, but Bob Leyse and I will summarize some important safety issues. Bob Leyse began working in the nuclear industry in 1950 and worked in nuclear safety at GE, Westinghouse, and EPRI. I am Mark Leyse, author of PRM-50-84, a petition accepted for consideration in NRC's rulemaking process for revisions to 50.46(b) and Appendix K to Part 50. I also wrote PRM-50-93.

PRM-50-93 is the subject of a user need request, dated April 26th, 2010, from Eric Leeds, Director, Office of Nuclear Reactor Regulations, to Brian Sheron, Director, Office of Nuclear Regulatory Research.

NRR's user need request states that, I cite extensive data from numerous multi-rod experiments and that their request is a high priority with a target due date of September 30, 2010.

In PRM-50-93, I argue that NRC's peak cladding temperature limit should be based on data from multi-rod Zircaloy severe fuel damage experiments, because such data demonstrates that the 2200-Fahrenheit limit is non-conservative. I also argue that the Baker-Just and Cathcart-Pawel equations are non-conservative for calculating the metal-water reaction rates that would occur in the event of a LOCA.[7] And I ask that a minimum reflood rate be specified.

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[7] Petitioner should have phrased this sentence as, "I also argue that the Baker-Just and Cathcart-Pawel equations are both non-conservative for use in analyses that predict the metal-water reaction rates that would occur in the event of a LOCA.

The page you have lists some of the multi-rod Zircaloy severe fuel damage experiments in which runaway oxidation commenced between 1832 and 2200 degrees Fahrenheit. It is reported that in the LOFT LP-FP-2 experiment—heated with actual decay heat—that runaway oxidation commenced at about 2060 degrees Fahrenheit.

In the Karlsruhe CORA program, there were about 20 experiments. A Karlsruhe paper states, "The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; that is, on bundle insulation. With a good bundle insulation in the CORA test facility, temperature escalation starts between 1100 and 1200 degrees Celsius."

And the page you have has a quote on single-rod quench experiments at Karlsruhe in which there were no temperature excursions during quenching, due to high radiative[8] heat losses.

Now Bob Leyse will discuss the Baker-Just equation.

Bob Leyse: I am Bob Leyse, author of denied PRM-50-76.

The licensing of ECCS in[9] many LWRs under Appendix K specifies Baker-Just. For emphasis, I repeat the licensing of ECCS in[10] many LWRs under Appendix K specifies Baker-Just.

In its technical analysis of PRM-50-76, the NRC fiercely defends Baker-Just. I quote, "The Baker-Just correlation using the current range of parameter inputs is conservative and adequate to assess Appendix K ECCS performance. Virtually every dataset published since the Baker-Just correlation was developed has clearly demonstrated the conservatism of the correlation above 1800 Fahrenheit." End of quote.

That is an interesting observation in light of data from Zircaloy multi-rod assembles that Mark Leyse has just cited. It is also revealing because the NRC did not even have access to the two key references in the Baker-Just report until April 2010. In response to my persistent demands, NRC acquired the documents and they were placed in ADAMS during April 2010. Short rods of half-inch-diameter Zircaloy 2 were induction heated underwater in Case 1, year 1954, and in steam in Case 2, year 1957.

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[8] The word "radiative" was transcribed as "radioactive" in the transcript, p. 187, line 21.
[9] The word "in" was transcribed as "and" in the transcript, on p. 188, line 1.
[10] The word "in" was transcribed as "and" in the transcript, on p. 188, line 3.

Shifting to[11] pages 7, 29, and 31 of the Commissioners' denial of PRM-50-76, I quote, "NRC's technical safety analysis demonstrates that current procedures for evaluating performance of ECCS are based on sound science and that no amendments to the NRC's regulations and guidance documents are necessary." End [of] quote.

Contrary to the Commissioners' observation, it is not sound science to combine the testing of single short rods of zirconium alloy with the testing of multi-rod stainless or Inconel assemblies in order to ascertain the performance of the emergency core cooling systems having thousands of zirconium alloy full-length rods.

Mark Leyse: An Oak Ridge National Laboratory paper discussing the CORA-16 experiment states, "The predicted and observed cladding thermal response are in excellent agreement until application of available Zircaloy oxidation kinetics models causes the low temperature 900 to 1200 degrees Celsius oxidation to be underpredicted."

And another ORNL paper states that, for the CORA-16 experiment, "cladding oxidation was not accurately predicted by available correlations." These papers are from the early 1990s, so the Baker-Just and Cathcart-Pawel equations were among the available Zircaloy oxidation kinetics models that under-predicted oxidation in the 1650-degree to 2200-degree Fahrenheit range.

Severe fuel damage experiments also show that eutectic reactions between fuel assembly components can commence below or at about 2200 degrees Fahrenheit; for example, the chemical reaction between Inconel spacer grids and Zircaloy fuel rods.

In its denial of PRM-50-76, in 2005, the NRC stated that more than 50 Zircaloy tests were conducted at the NRU reactor at Chalk River to evaluate the thermal hydraulic and mechanical deformation behavior of full-length bundles during a large-break LOCA, and that NRC is reviewing the data from that program to determine its value for assessing the current generation of codes such as TRACE. That was from 2005.

But almost all the Zircaloy heat-transfer tests conducted [at] Chalk River had peak cladding temperatures below 2000 degrees Fahrenheit. One test PCT was 2040 degrees Fahrenheit.

Except for the tests conducted at Chalk River, perhaps all [of] the main PWR and BWR heat-transfer experiments (after the original [FLECHT] tests) were conducted with

[11] The word "to" was transcribed as "from" in the transcript, on page 188, line 24.

stainless steel and Inconel 600 fuel rod simulators. Trying to relate this to what would occur in a LOCA in a reactor core with Zircaloy bundles simply does not work.

The NRC needs to conduct realistic heat-transfer experiments with multi-rod Zircaloy bundles in which the bundles would be heated up to at least 2200 degrees Fahrenheit.

The licensing basis PCTs of many plants do not provide necessary margins of safety. For example, the licensing basis PCT of Indian Point Unit 2 is 1937 degrees Fahrenheit, and Oyster Creek's is set at 2150 degrees Fahrenheit. Clearly, NRC's 2200-degree Fahrenheit PCT limit needs to be substantially lowered.

Thank you.

B. BWR Thermal Hydraulic Experiments and Core Spray Cooling

> There are, as you know, a number of problems in the BWR-FLECHT program. A great deal of this is resolved by the [General Electric] determination to prove out their ECC systems. ... Because the GE systems are marginally effective in arresting a thermal transient, there is little constructive effort on their part. ...the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven. From a licensing viewpoint, the effectiveness of top spray ECC has not been demonstrated nor has it been proven ineffective.[12]—J. W. McConnell

It seems that after the BWR-FLECHT program was concluded about forty years ago that there have not been any BWR heat transfer experiments conducted with parameters realistic enough to conclusively demonstrate that BWR core spray systems would be effective, in the event of a LOCA. Perhaps all of the primary BWR heat transfer experiments conducted after the BWR-FLECHT program was concluded were conducted with multi-rod Inconel 600 bundles.

So it seems that it has also never been conclusively demonstrated that BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs would effectively quench the fuel cladding in the event of LOCAs and prevent partial or complete meltdowns, if maximum cladding temperatures reached between 1832°F and 2200°F. This is highly problematic, because,

[12] J. W. McConnell, Aerojet internal memoranda; see Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-1, Union of Concerned Scientists, 1974, p. 5.11.

if a multi-rod Zircaloy bundle were heated up to maximum temperatures between 1832°F and 2150°F, it would (with high probability) incur autocatalytic oxidation. In the event of a LOCA, if autocatalytic oxidation occurred at a LWR, it would lead to a partial or complete meltdown.

Furthermore, to overcome the impression left from the BWR FLECHT program, BWR heat transfer experiments need to be conducted with multi-rod Zircaloy bundles, in which the bundles would be heated up to peak cladding temperatures of at least 2200°F. Such BWR heat transfer experiments need to be conducted in experiments modeling BWR/2, BWR/3, BWR/4, BWR/5, and BWR/6 ECCSs.

(It is noteworthy that there should be a regulation stipulating minimum allowable amounts of coolant to be supplied to each fuel bundle in the BWR core, in the event of a LOCA.[13])

1. Appendix K BWR Heat Transfer Coefficients

Appendix K to Part 50, ECCS Evaluation Models, I(D)(6), *Post-Blowdown Phenomena, Heat Removal by the ECCS, Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling*, states:

> Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7 x 7 fuel assembly array, the following convective coefficients are acceptable:
>
> a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.
>
> b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot °F^{-1}$ shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

[13] "Resolution of Generic Safety Issues: Item A-16: Steam Effects on BWR Core Spray Distribution" states that "to ensure the health and safety of the public, [BWR] core spray systems must supply a specified minimum amount of coolant to each fuel bundle in their respective reactor cores."

c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 $Btu \cdot hr^{-1} \cdot ft^{-2} \cdot °F^{-1}$ shall be applied to all fuel rods.

It is significant that Appendix K convective heat transfer coefficients for BWR Zircaloy fuel rods under spray cooling are based on data from the BWR Full Length Emergency Cooling Heat Transfer ("FLECHT") tests—from tests conducted with stainless steel electrically heated fuel rod simulators.

Regarding the fact that Appendix K heat transfer coefficients for BWR Zircaloy fuel rods are based on BWR FLECHT tests conducted with stainless steel electrically heated fuel rod simulators, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> *From the BWR FLECHT tests there is information on the heat transfer coefficients for both the convective heat flow to the water droplets and steam and for the reflood phase.* The FLECHT tests were made with an electrically heated mock-up of a 7 x 7 rod array complete with its channel box. *The convective heat transfer coefficients were determined from the residue of a thermal balance after all of the known inputs and outputs were calculated.* The factors considered were the electrical heat input, the rate of change of the heat content of the rods as calculated from their temperature history, and the calculated radiation from the rods to each other and to the channel walls. The residue from these inputs and outputs was ascribed to convective heat transfer. The convective heat transfer coefficients so determined could not be very accurate because their calculation involved taking the difference between two large numbers. The coefficients so obtained are small and are about what one would expect from the mechanisms of natural convection and radiation to steam (Exhibit 1113, p. 16-14).
>
> *The values of the calculated convective heat transfer coefficients depend to some extent upon the value used for the thermal emissivity of the stainless steel, since the convective heat transfer is obtained after subtracting the radiative heat transfer from the total.* Theoretically a high value of the emissivity leads to a low calculated convective heat transfer coefficient. Values of the emissivity measured after the tests ranged from 0.6 to 0.9 (Exhibit 461, p. 81 and Exhibit 1113, p. 16-14), and to add conservatism to the calculation, the Interim Policy Statement required the use of the highest measured emissivity, 0.9, for the calculation of the convective heat transfer coefficients. However, it turned out that this resulted in a higher coefficient (less conservative) for the critical inner rods, with a higher estimated standard error (Exhibit 461, Table 2). After reviewing the derivation of the coefficients as given in Exhibit 461, we

> believe that those originally listed as best estimates by General Electric are the most credible and should be used. The effect of this change on the peak cladding temperature will be small, about five degrees according to Exhibit 461.
>
> There has been a great deal of criticism of the BWR FLECHT tests, particularly by the Consolidated National Intervenors (Exhibit 1041, Chapter 5), and both General Electric and the Regulatory Staff have defended them (Closing Statements). However, for the purpose of calculating the maximum cladding temperature, only the derived heat transfer coefficients are of any great importance. The values obtained have always been known to have a high statistical error; furthermore, the values are low and reasonable, and there seems little to be gained by renewing the controversy over the manner of conducting and interpreting all features of the tests.
>
> The high but inevitable statistical error of the coefficients for the inner rods (1.5 ± 1.0 BTU/hr·ft^2·°F) is bothersome and leads to an estimated error band of as much as ±200°F in the calculated peak temperature in some circumstances (Exhibit 1113, p. 16-36). *The test bundle SS2N was used to derive the heat transfer coefficients*; another test bundle, SS4N, resulted in cladding temperatures 200°F higher than those of the bundle used as a standard; one half of this discrepancy could be explained by test differences, with the other half left to be attributed to statistical variations (Exhibit 1113, p. 16-38). The problem of these large statistical errors in the convective heat transfer coefficients is compensated to some extent by the fact that the coefficients were determined at atmospheric pressure, whereas the reactor would be at some elevated pressure at which the heat transfer would be improved (Exhibit 1113, p. 16-26).
>
> The evidence for the value 25 BTU/hr·ft^2·°F of the two phase reflooding heat transfer coefficient is sketchy (Exhibit 1032, p. II 6.3-51), but it is applied for only a short time because the high reflood rate would quickly quench the core, and the exact value is of little significance [emphasis added].[14]

So "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states that "[t]he

[14] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1125-1126. This document is located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML993200258; it is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

[BWR FLECHT] test bundle SS2N was used to derive the [Appendix K] heat transfer coefficients"[15] for BWR Zircaloy fuel rods.

(In the name "SS2N," "SS" stands for "stainless steel" and "N" stands for "Nichrome.")

And also regarding Appendix K heat transfer coefficients for BWRs, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states that "the heat transfer coefficients utilized in the GE core spray and reflood calculation model,[16] were derived on the basis of the SS2N test series."[17, 18]

In the BWR FLECHT SS2N test series, conducted from August to October 1969, three steady state tests were conducted with a peak power of 150 kW and coolant rates of 1.0-2.45 gallons/min.; 24 transient tests were conducted with peak powers of 100-250 kW, coolant rates of 2.45-5.0 gallons/min., and initial temperatures of 865-1850°F; eight combined spray and flooding tests were conducted with peak powers of 235-250 kW, coolant rates of 2.0-3.5 gallons/min. and 2.0-6.0 in./sec., and initial temperatures of 1335-1870°F.[19]

In the BWR FLECHT tests, five tests were conducted with Zircaloy electrically heated fuel rod simulators; however, Appendix K heat transfer coefficients for BWR Zircaloy fuel rods are not based on the data from the five Zircaloy tests.

Explaining the purpose of the five BWR FLECHT Zircaloy tests "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> [I]t was felt to be possible to evaluate heat transfer coefficients from [stainless steel] tests where the results would not be affected by [metal-water] reactions. The purpose of the [Zircaloy] tests was then to evaluate the validity of these assumptions by using [stainless steel] derived heat

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[15] *Id.*, p. 1126.

[16] J. D. Duncan and J. E. Leonard, "Thermal Response and Cladding Performance of an Internally Pressured, Zircaloy Cold, Simulated BWR Fuel Bundle Cooled by Spray Under Loss-of-Coolant Conditions," General Electric Co., San Jose, CA, GEAP-13112, April 1971, p. 58.

[17] Bruce C. Slifer, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Co., San Jose, CA, NEDO-10329, April 1971, p. 26.

[18] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," Environmental Quality Laboratory, California Institute of Technology, EQL Report No. 9, May 1975, p. A8-10.

[19] *Id.*, p. A8-5.

> transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.[20]

Discussing the PWR FLECHT tests, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> [T]he Commission sees no basis for concluding that the heat transfer mechanism is different for zircaloy and stainless steel, and believes that the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods.[21]

It is significant that the Atomic Energy Commission, also concluded that heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods in BWRs.

Regarding the problems with the heat transfer coefficients derived from the SS2N experiments with stainless steel fuel rod simulators, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> It seems probable that the difference between test and theory results from rigid adherence by GE to a time-dependant model of heat transfer coefficients which were derived from their SS2N tests and adopted as their "design model."[22] The design analysis method, based on the SS2N time history, apparently did not permit accommodation of the idiosyncrasies of the Zr2K test experience with its rod heater failures and [thermocouple] equipment malfunctions. Consequently, the predicted results might not reasonably be expected to correspond well with the reality of the Zr2K test. Whether or not design basis production of LOCA thermal histories would agree well with an actual transient also remains to be shown. Results imply that the GE thermal analysis method may be a weak predictive tool and more effort appears to be needed in model development. However, it does appear that with sufficient analysis, FLECHT results would be adequate to form a basis for demonstrating the development of conservative analytical design methods.[23]

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[20] *Id.*, p. A8-7.

[21] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1124. This document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

[22] Bruce C. Slifer, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," General Electric Co., San Jose, CA, NEDO-10329, April 1971, p. 26.

[23] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," EQL Report No. 9, pp. A8-27, A8-28.

2. Appendix K BWR Heat Transfer Coefficients for New BWR Fuel Assembly Designs

It is significant that Appendix K specifies that its BWR heat transfer coefficients are to be used for fuel rods in a 7 x 7 fuel assembly array. Since Appendix K was written, new BWR fuel assembly designs have come into use, so Appendix K BWR heat transfer coefficients have been converted so that they can also apply to new BWR fuel assembly designs.

Discussing the application of heat transfer coefficients to various BWR fuel assembly designs, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel" states:

> Although the channel size has not changed significantly since the 1970's, the BWR fuel assembly designs have changed in many ways. These changes have resulted in a larger number of smaller diameter fuel rods as well as various non-boiling water channel designs. ... Spray heat transfer tests have been performed (*e.g.*, the BWR FLECHT test program) from which convective spray heat transfer coefficients have been derived. CENPD-283-P-A...provides a summary of these tests *and describes how the spray cooling heat transfer coefficients are applied to various fuel geometries*. ... The BWR FLECHT tests, which simulated a 7x7 array, showed that the convective coefficients are dependant on the location of the fuel rod relative to its proximity to the channel enclosure (corner rod, outer row rod, or interior rod). Table 6-2 lists the heat transfer coefficients that are acceptable for use in an Appendix K analysis of 7x7 fuel [emphasis added].[24]

(Table 6-2, Appendix K Spray Cooling Heat Transfer Coefficients, states that the values for heat transfer coefficients are: for corner rods—17.0 $W/m^2 \cdot K$, for side rods—19.9 $W/m^2 \cdot K$, for inner rods—8.5 $W/m^2 \cdot K$, and for channel—28.4 $W/m^2 \cdot K$.[25])

It is significant that BWR FLECHT spray heat transfer coefficients for 7x7 fuel assembly arrays have been converted so that they can be used for 8x8 fuel assembly arrays.[26] It certainly stands to reason that BWR FLECHT spray heat transfer coefficients

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[24] John A. Blaisdell, Westinghouse, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel," WCAP-16078-NP-A, November 2004, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050390435, p. 30.
[25] *Id.*, p. 31.
[26] *Id.*

for 7x7 fuel assembly arrays have also been converted so that they can be used for 9x9 and 10x10 fuel assembly arrays.

3. Criticisms of the BWR FLECHT Tests

Discussing one of Henry Kendall and Daniel Ford's, of Consolidated National Intervenors ("CNI"),[27] criticisms of the BWR-FLECHT tests, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The first complaint [regarding the BWR-FLECHT tests] was that although all BWR fuel rods are manufactured of a zirconium...alloy, Zircaloy, only 5 of the 143 FLECHT tests utilized [Zircaloy] rods. The remaining 138 tests were conducted with stainless steel...rods. *Since...[Zircaloy] reacts exothermically with water at elevated temperatures, contributing additional energy to that of the decaying fission products, the application of water to the core has the potential of increasing the heat input to the fuel rods rather than cooling them, as desired.* The small number of [Zircaloy] tests in comparison with the total test program was seriously faulted by the CNI [emphasis added].[28]

And discussing the use of stainless steel heater-rod assemblies in the FLECHT program, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The [stainless steel] rods were apparently chosen primarily for their durability. They could be used repeatedly in testing (for 30 or 40 individual tests) without substantial changes in response over the series. ...
>
> On the other hand, *as a result of metal-water reactions, [Zircaloy] rods could be used only once* and then had to be subjected to a destructive post-mortem examination after the test [emphasis added].[29]

(It is noteworthy that, regarding the oxidation reactions of stainless steel and Zircaloy, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI" states that "[t]he rate of [stainless] steel oxidation is small relative to the oxidation of Zircaloy at temperatures below 1400°K. At higher temperatures and near

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[27] Henry Kendall and Daniel Ford of Union of Concerned Scientists were the principal technical spokesmen of Consolidated National Intervenors, in the AEC ECCS rulemaking hearing.
[28] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," EQL Report No. 9, pp. A8-2, A8-6.
[29] *Id.*, p. A8-6.

the [stainless] steel melting point, the rate of [stainless] steel oxidation exceeds that of Zircaloy;"[30] and states that "the rate of reaction for [stainless] steel exceeds that of Zircaloy above 1425°K. *The heat of reaction, however, is about one-tenth that of Zircaloy, for a given mass gain*" [emphasis added].[31])

And regarding Aerojet internal memoranda that provide commentary on the BWR-FLECHT program consistent with that presented by CNI, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> [Aerojet] internal memoranda provide commentary on the BWR-FLECHT program quite consistent with that presented by CNI. Thus, for example, J. W. McConnell (who will be co-author, with Dr. Griebe, of the as-yet-unpublished BWR-FLECHT final report from [Aerojet]) wrote:
>
> "There are, as you know, a number of problems in the BWR-FLECHT program. A great deal of this is resolved by the GE determination to prove out their ECC systems. Their role in this program can only be described as a conflict of interest as is the Westinghouse portion of PWR-FLECHT. Because the GE systems are marginally effective in arresting a thermal transient, there is little constructive effort on their part. ... A combination of poor data acquisition and transmission, faulty test approaches (probably caused by crude test facilities) and the marginal nature of these tests has produced a large amount of questionable data. It appears probable that the results of these tests can be interpreted. *But the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven.* From a licensing viewpoint, the effectiveness of top spray ECC has not been demonstrated nor has it been proven ineffective [emphasis added]."[32]

So J. W. McConnell concluded that "the ability to predict accurately the heat transfer coefficient and metal-water reactions may not be proven."[33]

Discussing the concept of separating the zirconium-water reaction from cladding heat transfer mechanisms, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> [Another] reason for using more [stainless steel] than [Zircaloy] rods involves the problems of simplifying heat transfer analyses by separating

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[30] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 2.2.
[31] *Id.*, p. 4.4.
[32] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 5.11.
[33] *Id.*

> the [metal-water] reaction from the physical processes of cooling rods which were not undergoing [a metal-water] reaction. *It was assumed that the [metal-water] reaction was an independent heat input mechanism to the fuel rods, separable from the basic heat transfer processes of cooling.* On this basis, the [stainless steel] rods permitted direct determination of the applicable heat transfer coefficients for the cooling mechanisms without supplementary heat input complications. *The validity of this concept of separability of the two heat transfer mechanisms rests on the assumption that the radiative and convective heat transfer processes for heat transmission between fuel rods and the coolant fluid are essentially independent of the fuel rod materials, and thus are functions primarily only of temperature and fluid flow conditions. Thus, it was felt to be possible to evaluate heat transfer coefficients from [stainless steel] tests where the results would not be affected by [metal-water] reactions.* The purpose of the [Zircaloy] tests was then to evaluate the validity of these assumptions by using [stainless steel] derived heat transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.
>
> The weakness of these arguments for rod material selection is that because of the small number of [Zircaloy] tests and the poor quality of the [Zircaloy] results, *questions remain concerning the validity of the assumptions of the equivalence of non-reactive heat transfer characteristics for the two materials and the legitimacy of decoupling the metal-water reaction from the clad heat transfer mechanisms* [emphasis added].[34]

(It is significant that in the ECCS rulemaking hearing, the Atomic Energy Commission ("AEC") Commissioners did not seem concerned about decoupling the zirconium-water reaction from cladding heat transfer mechanisms. The AEC Commissioners merely concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients. Regarding the AEC Commissioners' conclusion, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The reasonable conclusion was reached that the effect of the difference between Zircaloy and stainless steel, if any, would be small. There is a difference, of course, in the rate of heat generation from steam oxidation, *but this heat is deposited within the metal under the surface of the oxide film. The presence of this heat source should not affect the heat transfer*

[34] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-7.

coefficients, which depend on conditions in the coolant outside the rod [emphasis added].[35]

So the AEC Commissioners concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients, maintaining that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside fuel rods. (Petitioner discusses the fallacy of the AEC Commissioners' conclusion in the following section.))

It is significant that J. W. McConnell concluded that "from a licensing viewpoint, the effectiveness of top spray ECC has not been demonstrated nor has it been proven ineffective"[36] in the BWR-FLECHT program.

4. The Fallacy of the AEC Commissioners' Conclusion that the Heat Generated from the Exothermic Zirconium-Water Reaction would Not Affect the Coolant Outside Fuel Rods

To discuss the fallacy of the AEC Commissioners' conclusion that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside fuel rods, Petitioner will discuss PWR FLECHT Run 9573. Run 9573 was one of the four tests conducted with Zircaloy cladding in the PWR FLECHT test program; the assembly used in run 9573 incurred autocatalytic (runaway) oxidation.

Run 9573 was part of the PWR FLECHT test program; however, the exothermic zirconium-water reaction that occurred in the test is pertinent to both PWR and BWR Zircaloy fuel rods in LOCA environments. It is significant that a Zircaloy assembly used in the BWR FLECHT program—the Zr2K test assembly—also incurred autocatalytic oxidation. (The BWR FLECHT Zr2K test is discussed in the following section.)

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[35] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1123-1124; this document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

[36] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, p. 5.11.

It is significant that "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> [T]he Commission sees no basis for concluding that the heat transfer mechanism is different for zircaloy and stainless steel, and believes that the heat transfer correlations derived from stainless steel clad heater rods are suitable for use with zircaloy clad fuel rods. It is apparent, however, that more experiments with zircaloy cladding are needed to overcome the impression left from run 9573."[37]

According to the NRC, "[t]he 'impression [left from FLECHT run 9573]' referred to by the AEC Commissioners in 1973, appears to be the fact that run 9573 indicates lower 'measured' heat transfer coefficients than the other three Zircaloy clad tests reported in ["PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report"] when compared to the equivalent stainless steel tests."[38] The NRC also stated, regarding the results of FLECHT run 9573, that the AEC Commissioners were not "concern[ed] about the zirconium-water reaction models."[39]

Discussing the concept of separating the zirconium-water reaction from cladding heat transfer mechanisms, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors" states:

> The second reason for using more [stainless steel] than [Zircaloy] rods involves the problems of simplifying heat transfer analyses by separating the [metal-water] reaction from the physical processes of cooling rods which were not undergoing [a metal-water] reaction. *It was assumed that the [metal-water] reaction was an independent heat input mechanism to the fuel rods, separable from the basic heat transfer processes of cooling.* On this basis, the [stainless steel] rods permitted direct determination of the applicable heat transfer coefficients for the cooling mechanisms without supplementary heat input complications. *The validity of this concept of separability of the two heat transfer mechanisms rests on the assumption that the radiative and convective heat transfer processes for*

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[37] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, p. 1124; this document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

[38] NRC, "Denial of Petition for Rulemaking (PRM-50-76)," June 29, 2005, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML050250359, pp. 16-17.

[39] *Id.*, p. 17.

heat transmission between fuel rods and the coolant fluid are essentially independent of the fuel rod materials, and thus are functions primarily only of temperature and fluid flow conditions. Thus, it was felt to be possible to evaluate heat transfer coefficients from [stainless steel] tests where the results would not be affected by [metal-water] reactions. The purpose of the [Zircaloy] tests was then to evaluate the validity of these assumptions by using [stainless steel] derived heat transfer coefficients to evaluate (or provide post-test predictions) of the thermal response of [Zircaloy] bundles.

The weakness of these arguments for rod material selection is that because of the small number of [Zircaloy] tests and the poor quality of the [Zircaloy] results, questions remain concerning the validity of the assumptions of the equivalence of non-reactive heat transfer characteristics for the two materials and the legitimacy of decoupling the metal-water reaction from the clad heat transfer mechanisms [emphasis added].[40]

And opining on the concept of separating the zirconium-water reaction from cladding heat transfer mechanisms, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors" states:

> The reasonable conclusion was reached that the effect of the difference between Zircaloy and stainless steel, if any, would be small. There is a difference, of course, in the rate of heat generation from steam oxidation, but this heat is deposited within the metal under the surface of the oxide film. *The presence of this heat source should not affect the heat transfer coefficients, which depend on conditions in the coolant outside the rod.*[41]

So the AEC Commissioners concluded that the heat generated from the exothermic zirconium-water reaction would not affect heat transfer coefficients, maintaining that the heat generated from the exothermic zirconium-water reaction would not affect the coolant outside the rod.

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[40] Fred C. Finlayson, "Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors," p. A8-7.

[41] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, pp. 1123-1124; this document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

It is significant that within the first 18.2 seconds of FLECHT run 9573,[42] "negative heat transfer coefficients were observed at the bundle midplane for 5...thermocouples;"[43] *i.e.*, more heat was transferred into the bundle midplane than was removed from that location. In petition for rulemaking 50-76 ("PRM-50-76"), Robert H. Leyse, the principal engineer in charge of directing the Zircaloy FLECHT tests and one of the authors of "PWR FLECHT Final Report," states that "[t]he negative heat transfer coefficients [occurring within the first 18.2 seconds of run 9573] were calculated as a result of a heat transfer condition during which more heat was being transferred into the heater than was being removed from the heater[; used in the FLECHT tests to simulate fuel rods]. And the reason for that condition was that the heat generated from Zircaloy-water reactions at the surface of the heater added significantly to the linear heat generation rate at the location of the midplane thermocouples."[44]

So the heat generated from the exothermic oxidation reaction of the Zircaloy cladding (and Zircaloy spacer grids) was transferred from the cladding's reacting surface inward. Indeed, the Zircaloy-cladding heater rods were very hot internally, where the thermocouples were located; yet, nonetheless, the heater rods became a heat sink.[45]

Additionally, the exothermic oxidation reaction of the Zircaloy heated a mixture of steam and hydrogen, and entrained water droplets. Westinghouse agrees with this claim; in its comments regarding PRM-50-76, Westinghouse stated, "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam. The reaction would have occurred at the hot spots on the heater rods, on the Zircaloy guide tubes, spacer grids, and steam probe."[46]

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[42] F. F. Cadek, D. P. Dominicis, R. H. Leyse, Westinghouse Electric Corporation, WCAP-7665, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report," April 1971, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML070780083, p. 3-97.

[43] *Id.*, p. 3-98.

[44] Robert H. Leyse, "PRM-50-76," May 1, 2002, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022240009, p. 6.

[45] Robert H. Leyse, "Nuclear Power Blog," August 27, 2008; located at: http://nuclearpowerblog.blogspot.com.

[46] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," October 22, 2002, located at:

Regarding steam temperatures measured by the seven-foot steam probe, "PWR FLECHT Final Report" states:

> At the time of the initial [heater element] failures, midplane clad temperatures were in the range of 2200-2300°F. The only prior indication of excessive temperatures was provided by the 7 ft steam probe, which exceeded 2500°F at 16 seconds (2 seconds prior to start of heater element failure).[47]

Therefore, it is reasonable to conclude that a superheated mixture of steam and hydrogen, and entrained water droplets, caused heating of Zircaloy cladding in the midplane location of the fuel rod. It is also reasonable to conclude that the "negative heat transfer coefficients [that] were observed at the bundle midplane for 5...thermocouples"[48]—the occurrence of more heat being transferred into the bundle midplane than was removed from that location—within the first 18.2 seconds of FLECHT run 9573, were caused by an exothermic zirconium-water reaction. Additionally, it is reasonable to conclude that "the impression left from [FLECHT] run 9573" cannot be separated from concerns about zirconium-water reaction models.

Furthermore, because, as Westinghouse stated, "[t]he high fluid temperature [that occurred during FLECHT run 9573] was a result of the exothermic reaction between the zirconium and the steam,"[49] the AEC Commissioners' conclusion that "the presence of...heat [generated from the exothermic zirconium-water reaction] should not affect...heat transfer coefficients, which depend on conditions in the coolant outside the rod"[50] is erroneous. Clearly, the exothermic zirconium-water reaction affects the coolant outside the cladding by heating a mixture of steam and hydrogen, and entrained water droplets; therefore, the zirconium-water reaction cannot legitimately be separated from cladding heat transfer mechanisms.

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www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML022970410, Attachment, p. 3.

[47] F. F. Cadek, D. P. Dominicis, R. H. Leyse, "PWR FLECHT Final Report," p. 3-97.

[48] *Id.*, p. 3-98.

[49] H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse, "Comments of Westinghouse Electric Company regarding PRM-50-76," Attachment, p. 3.

[50] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," p. 1124; this document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50."

5. More Criticisms of the BWR FLECHT Tests

Regarding the BWR-FLECHT Program, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing,"—section V.G.2., CNI Guidance for Commission on Information Needs for LOCA Analysis, FLECHT Program, BWR-FLECHT—states:

> The BWR-FLECHT Program was carried out by the General Electric Company (GE) under a subcontract of the Idaho Nuclear Corporation, itself a contractor to the test program sponsor, the AEC. Roger Griebe of [Aerojet] coordinated the program, as project engineer, for [Aerojet], GE and the AEC. The UCS devoted a very substantial effort to an independent analysis of the BWR-FLECHT Program and its weaknesses. To CNI's knowledge this has been the only independent review of the program which has been carried out and made available in the public literature. CNI believes that the program failures and weaknesses that are identified in CNI testimony (Exhibit 1041) are overwhelmingly supported by the testimony of the knowledgeable engineers in the AEC contract laboratories who were associated and familiar with the elements of the program. CNI testimony sets out the case in substantial detail. In brief, the program was characterized by narrow scope, limited range of parameters investigated (many inappropriate to the tasks at hand), the use of incorrect materials, crude and incompetent instrumentation and operating techniques (with consequent major equipment malfunctions), and, as a culminating weakness, expansive and overgenerous interpretations. These latter, in CNI's view, misrepresented badly technical information available from the test results. In particular, in a test series of over 150 tests only one, ZR-2, simulated fuel rod swelling and rupture and the associated channel blockage which would be expected to occur under LOCA circumstances. It was a unique test, a circumstance which should not have occurred, and was highly defective. The importance of test ZR-2 was reflected in the hearing record in the extensive time taken by participants to discuss and to try to illuminate the nature and sources of the test weaknesses and to determine reliability what the test had to say.
>
> J.O. Zane of [Aerojet] did not believe ZR-2 was a demonstration of the ability of BWR ECCS to operate in a LOCA (Tr. 6415-6423).
>
> C.G. Lawson of ORNL said test ZR-2 was borderline and more tests were required employing pressurized fuel rods as in ZR-2 (Tr. 5719-5725).
>
> P.L. Rittenhouse of ORNL stated that it was unreasonable and arbitrary to conclude that test ZR-2 shows flow blockage would not inhibit the spray cooling system (Tr. 4757).

Roger Griebe, the engineer at [Aerojet] with perhaps the most familiarity with the BWR-FLECHT Program, said that General Electric did not have the enthusiasm he felt necessary to conduct the tests (Tr. 6935-6945) and that he could not personally defend the General Electric conclusions (Tr. 7006). He said that GE "overstated" points, became carried away with impressions not verified by the technical data, and that the General Electric conclusion that protection was provided by the ECCS against all break sizes was not completely supported in the FLECHT data (Tr. 7100 et. seq.). In cross-examination he stated that he felt the GE reporting of the data was "tremendously slanted" (Tr. 7117). [Aerojet]-GE-AEC internal memorandum released by CNI bearing on the conduct of the BWR-FLECHT tests tells an even more dismal story of the conduct and interpretation of these tests than is contained either in CNI testimony or in the oral transcript. Based on the careful reading of the memorandum in the light of CNI's analysis of the tests and of the cross-examination of [Aerojet] and GE witnesses, CNI has concluded that in effect GE tried to approach elements of the test program, and attempted to interpret the results, in ways wholly inconsistent with the technical content of the test program.

These [Aerojet] memos, incorporated in Exhibit 1153, note:

"This was not [a] satisfactory demonstration test—the same need exists today—in fact, the need is greater because margin appears to be less than originally expected."

"[GE's] role in this program can only be described as a conflict of interest... Because the GE systems are marginally effective...there is little constructive effort on their part. ... A combination of poor data acquisition and transmission, faulty test approaches (probably caused by crude test facilities) and the marginal nature of these tests has produced a large amount of questionable data."

"...the close coupling between GE-FLECHT Project Group [the testing group] and GE licensing group has precluded pursuing a completely objective experimental program in an expedient manner."

An internal investigation of the failure of some of the GE design test apparatus to function properly, concluded:

"...the 'why' of the situation has come down to the simple fact that we believed GE was doing the job for which they were paid... the GE effort in heater development has been demonstrated to be seriously inadequate."

CNI's conclusion has been that it has proven inappropriate and damaging for the AEC to have established a policy of letting industry do the testing

to check out the industry's own claims regarding safety system performance. The inherent conflict of interest has led to a testing program of narrow scope and poor quality.

One should note the letter of June 30, 1970 (Exhibit 1029) from William B. Cottrell, Director of the ORNL Nuclear Safety Program. In this letter to A.J. Pressesky:

"The Commission's position in its support of nuclear safety research is *seriously compromised* by relegating significant portions of the nuclear safety research and development program to the same industry it would license [emphasis added]."

Later in the letter, Cottrell cites examples known to him wherein a reactor vendor when given the responsibility for undertaking safety research on the reactor he was selling failed to get to the heart of the safety problems in question: In regard to fuel rod swelling in LOCA circumstances, the reactor vendors, on the basis of their own in-house R&D, concluded that the diametrical swelling of the fuel rods during the LOCA would be less than 30%. This they later increased to 60%. ORNL experiments subsequently demonstrated that swelling greater than 100% was possible under realistic conditions over significant portions of the core. Additionally, vendors maintained that embrittlement would not occur in the LOCA and hence its consequences need not be considered in evaluating the accident. Cottrell noted that ORNL experiments have been much more pessimistic in this regard.

With regard to [the] BWR-FLECHT Program, CNI concluded that in effect *GE tried to sabotage elements of the program* and attempted to interpret the results in ways utterly inconsistent with the test program's technical content. It is CNI's conclusion that the judgments set forth in its testimony are now even more powerfully supported by the hearing record [emphasis not added]. No recovery from the defects in the BWR-FLECHT Program are possible without a new program of greater scope being planned and carried out, like a new PWR-FLECHT Program, carried out in a way essentially free of the conflicts of interest that so seriously undermined the FLECHT programs since their inception.[51]

[51] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-l, pp. 5.37-5.41.

(It is noteworthy that despite the testimony of a number of safety experts that ZR-2 did not demonstrate the ability of BWR ECCS to work effectively, the AEC Commissioners concluded:

> [T]he data of the Zr-2 BWR FLECHT experiment were cited as *evidence for the effectiveness of spray cooling*, although no temperature measurements were made at the positions of maximum bulging. We believe that additional assessments need to be made of these effects.
>
> In addition to the primary heat transfer effects of taking into consideration the swelling and rupture of the cladding, there would be important secondary effects arising from the steam oxidation of the cladding by the steam. Higher temperatures would lead to increased oxidation, which would contribute to a further increase in temperature, and the opening in the cladding would allow oxidation on the inside, again increasing the calculated temperature [emphasis added].[52])

6. Criticisms of GE's BWR Core Spray Tests

Regarding problems with BWR core spray cooling, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing"—section V.I., CNI Guidance for Commission on Information Needs for LOCA Analysis, BWR Core Spray—states:

> CNI identified the weaknesses in simulations of BWR core spray cooling of a full-length bundle as implemented in GE's uniquely defective test ZR-2 (Exhibit 1041, p. 5.39). CNI raised the possible existence of core spray diversion mechanisms that could in a BWR LOCA result in spray starvation of the central and hotter fuel bundles of a core. Similar concerns have been raised by Aerojet (Exhibit 1032, p. 124). GE diligence in resolving these concerns leaves a substantial amount to be desired. Cross-examination established (Tr. 13,925 et. seq.) that GE had done *no experiments* to determine spray droplet size distribution *either* at spray nozzles or at bundle tops (Tr. 13,953-13,956). They have done *no experiments* to determine the degree of superheat of the ejected steam. However, they "assume" the steam is saturated. They have done *no experiments* to determine steam temperature at bundle exit *or* to determine steam velocity at the bundle top. Moreover, the GE analytical model *does not furnish* a prediction of the velocity *nor compute entrainment* of spray

[52] Dixy Lee Ray, Clarence E. Larson, William O. Doub, William E. Kriegsman, William A. Anders, "Commission Decision on Rulemaking for Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors," CLI-73-39, 6 AEC 1085, December 28, 1973, p. 1106; this document is Attachment 3 to "Documents Related to Revision of Appendix K, 10 CFR Part 50," September 23, 1999.

drops by upward streaming steam. Since GE believes superheating "will not occur in the reactor" they carried out no calculations to see if superheating results in velocity increases. GE, however, stated that in some of the ZR FLECHT tests a thermocouple was placed in the exit flue (steam exhaust line) some distance from the bundle exit. Temperatures at that point would surely be lower than at the bundle top. The thermocouple showed temperatures [of up] to 250°F. This information effectively invalidates the GE statement that no experimental information contradicted the saturation assumption especially in view of their not setting forth experimental results confirming the assumption. A hint that the exit steam velocities may be very great is given by Roger Griebe's observations that the steam plume and apparent steam exit velocity from test ZR-3 and 4 were unexpectedly and uniquely large (Staff reference 16.20 in the Regulatory Staff Supplementary Testimony). These observations raise some unresolved questions about the applicability of the GE core spray tests which are discussed next.

GE carried out spray tests using upward *air*flow, *with no heating*, to attempt to simulate spray operation (Tr. 13,919-13,925), but in view of the remarks above, CNI believes the test results to be inapplicable. Lawson of ORNL has criticized the tests because of the difference between steam and air on droplet entrainment (Tr. 5790-5795).

CNI shares the Aerojet view that spray may not work (Exhibit 1032). CNI believes that GE has not made adequate attempts to establish a contrary view, a situation which may reflect the fact that the contrary may not be supportable. The GE attitude toward test conduct and the interpretation of test data is well established with respect to the BWR-FLECHT tests by the [Aerojet]-GE-AEC internal memoranda placed in the record by CNI. These are discussed in [the FLECHT Section], above. It is shown that GE made a poor accommodation to the conflict of interest inherent in their assumption of responsibility for carrying out the tests. GE's diligence in core spray effectiveness tests is no better than in FLECHT, and their conclusions no better supported. As pointed out in [Section VII, "The Regulatory Failure,"] there is no assurance available from the FLECHT program that a BWR bundle can be cooled successfully at the spray rates employed in the tests and with the identified weaknesses in spray injection simulation (Exhibit 1041, p. 5.39) that reduced the conservatism of the test. Spray diversion in a BWR LOCA would reduce even further the controllability of the accident and it is without question a possibility which has not been eliminated. It requires prompt resolution.[53]

[53] Daniel F. Ford and Henry W. Kendall, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing," AEC Docket RM-50-1, pp. 5.43-5.45.

Regarding core spray distribution, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> Another example relates to the distribution of core spray assumed by GE. GE performed tests involving air up-flow with non-heated simulated bundles as the basis for its core spray distribution assumptions. In June of 1972, the Regulatory Staff ask[ed] GE to describe the basis upon which it could conclude that these air up-flow tests are applicable to the reactor situation. These tests were performed many years ago [before 1973] by GE and they have been the basis upon which GE boiling water reactor emergency core cooling systems have been evaluated for several years, and they are the basis upon which the model approved by the Regulatory Staff in June 1971 determines now much emergency cooling water is delivered to the core. In asking this question, the Regulatory Staff raise[d] the most fundamental doubt about the kind of review that it made of the GE LOCA analysis during all [of] these years in which it has been allowing GE reactors to operate. [54]

7. More Recent BWR Thermal Hydraulic Experiments have been Conducted with Inconel 600 Fuel Rod Simulators

Regarding the prospect of planning and conducting a new BWR-FLECHT program, "An Assessment of the Emergency Core Cooling Systems Rulemaking Hearing" states:

> No recovery from the defects in the BWR-FLECHT Program are possible without a new program of greater scope being planned and carried out, like a new PWR-FLECHT Program, carried out in a way essentially free of the conflicts of interest that so seriously undermined the FLECHT programs since their inception.[55]

Petitioner would add that such a new BWR-FLECHT program would have to be conducted with Zircaloy fuel assemblies. It would also be necessary that the PCTs of such tests exceeded those of the PWR Thermal-Hydraulic Experiment 1 ("TH-1") tests, conducted at Chalk River in the early '80s, where the test planners—"for safety

[54] *Id.*, pp. 7.5-7.6.
[55] *Id.*, p. 5.41.

purposes"—did not want the maximum PCTs of the TH-1 tests to exceed 1900°F[56]—300°F below the 10 C.F.R. § 50.46(b)(1) PCT limit of 2200°F.

Unfortunately, it seems that none of the primary BWR heat-transfer experiments performed since the BWR-FLECHT tests were conducted with Zircaloy fuel assemblies.

Perhaps all of the primary BWR heat-transfer experiments performed since the BWR-FLECHT tests were conducted with Inconel 600 fuel rod simulators. For example, the Two-Loop Test Apparatus ("TLTA") facility had electrically heated Inconel 600 fuel rod simulators,[57] the Rig of Safety Assessment ("ROSA") III facility had electrically heated Inconel 600 fuel rod simulators,[58] and the Full Integral Simulation Test ("FIST") facility had electrically heated Inconel 600 fuel rod simulators.[59]

Additionally, the BWR FIX-II test facility had electrically heated Inconel 600 fuel rod simulators[60] and the NUPEC BWR Full-Size Fine-Mesh Bundle Test ("BFBT") facility had electrically heated Inconel 600 fuel rod simulators.[61]

Petitioner has not been able to locate information identifying the cladding material that was used in the fuel rod simulators in the 30° Steam Sector Test Facility ("SSTF"); in the SSTF, it is doubtful that Zircaloy was used as the fuel rod simulator

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[56] 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, 1981, located in ADAMS Public Legacy, Accession Number: 8104140024, p. 3-3.

[57] GE Nuclear Energy, "Licensing Topical Report: TRACG Qualification," NEDO-32177, Revision 3, August 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072480029, p. 5-27.

[58] Y. Koizumi, M. Iriko, T. Yonomoto, K. Tasaka, "Experimental Analysis of the Power Curve Sensitivity Test Series at ROSA-III," Nuclear Engineering and Design, 86, 1985, pp. 268, 270.

[59] General Electric, "BWR Full Integral Simulation Test (FIST) Program Facility Description Report" NUREG/CR-2576, EPRI NP-2314, GEAP-22054, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML073461126, pp. 2-32, 2-37; and Siemens, "EXEM BWR-2000 ECCS Evaluation Model," EMF-2361 (NP), October 2000, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML003772936, p. 5-2.

[60] GE Nuclear Energy, "Licensing Topical Report: TRACG Qualification," NEDO-32177, Revision 3, August 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072480029, pp. 5-119, 5-129.

[61] B. Neykov, F. Aydogan, L. Hochreiter, K. Ivanov, H. Utsuno, K. Fumio, E. Sartori, "NUPEC BWR Full-Size Fine-Mesh Bundle Test (BFBT) Benchmark," Volume I: Specifications, NEA/NSC/DOC(2005)5, June 2005, pp. 15, 34.

cladding material. The SSTF experiments used steam injection to simulate core heat[62] the maximum temperature of the steam was 800 F.[63]

(It is noteworthy that many of the papers reporting on BWR heat-transfer experiments do not mention what type of cladding material was used in the fuel rod simulators in the experiments they describe. For example, Petitioner has not located any papers that state what type of cladding material is used in the fuel rod simulators in the Purdue University Multidimensional Integral Test Assembly ("PUMA") facility. Most likely, the PUMA facility—currently investigating BWR-related problems—uses Inconel 600 fuel rod simulators: the Rod Bundle Heat Transfer ("RBHT") facility at Penn State University—currently investigating PWR-related problems—uses Inconel 600 fuel rod simulators.[64] Also, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, which describes many experimental facilities, does not mention what type of cladding material was used in the fuel rod simulators at the experimental facilities it describes.)

It is significant that Inconel 600 does not oxidize nearly as much as Zircaloy in the design-basis accident temperature range.

Discussing Inconel 600's resistance to oxidation, "INCONEL alloy 600," states:

> INCONEL alloy 600 is widely used in the furnace and heat-treating fields for retorts, boxes, muffles, wire belts, roller hearths, and similar parts which require resistance to oxidation and to furnace atmospheres. ... The alloy's resistance to oxidation and scaling at 1800°F (980°C) is shown in Figure 11.[65]

Figure 11 of "INCONEL alloy 600," depicts a graph of the results of cyclic oxidation tests at 1800°F (980°C), in which there were alternating intervals of 15 minutes of heating and 5 minutes of cooling in air: Inconel 600 oxidized less than stainless steel

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[62] NRC, "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053490333, p. 6.5-11.

[63] NRC, (Appendix A) "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, 1988, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML053620415, Appendix A, p. A-208.

[64] Donald R. Todd, Cesare Frepoli, Lawrence E. Hochreiter, "Development of a COBRA-TF Model for the Penn State University Rod Bundle Heat Transfer Program," 7th International Conference on Nuclear Engineering, Tokyo, Japan, April 19-23, 1999, ICONE-7827, p. 3.

[65] Special Metals Corporation, "INCONEL alloy 600," www.specialmetals.com, SMC-027, 2008, p. 11.

(type 304), stainless steel (type 309), and Inconel 800HT. Inconel 600 oxidized very little over a period of 1000 hours of cyclic exposure time.

Additionally, in an Advisory Committee on Reactor Safeguards, subcommittee meeting on thermal hydraulic phenomena, on July 7, 2008, a participant, Mr. Kelly, discussing LOCA phenomena, stated that Inconel has "almost no oxidation."[66]

Henry Kendall and Daniel Ford's criticisms of the BWR FLECHT tests conducted with stainless steel fuel rod simulators would also apply to BWR thermal hydraulic experiments conducted since the early 1970s with Inconel 600 fuel rod simulators. To conclusively demonstrate that BWR ECCSs would be effective, in the event of a LOCA, it would be necessary to conduct BWR heat transfer experiments with multi-rod Zircaloy bundles, in which the bundles would be heated up to peak cladding temperatures of at least 2200°F. Experiments with Inconel 600 fuel rod simulators are inadequate.

Furthermore, interpretations of the results of experiments conducted with Inconel 600 fuel rod simulators would most likely lead the interpreters to false conclusions. For example, a multi-rod Inconel 600 bundle heated up to peak cladding temperatures between 1832°F and 2200°F would not incur autocatalytic oxidation; however, a multi-rod Zircaloy bundle heated up to peak cladding temperatures between 1832°F and 2200°F would (with high probability) incur autocatalytic oxidation.

[66] Mr. Kelly, NRC, Advisory Committee on Reactor Safeguards, Transcript of Subcommittee Meeting on Thermal Hydraulic Phenomena, July 7, 2008, p. 168.

III. CONCLUSION

If implemented, the regulations proposed in PRM-50-93 and PRM-50-95 would help improve public and plant-worker safety.

Respectfully submitted,

Mark Edward Leyse
P.O. Box 1314
New York, NY 10025
markleyse@gmail.com

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