PRM-50-105

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RULEMAKINGS AND
ADJUDICATIONS STAFF

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

Attention: Rulemakings and Adjudications Staff

PETITION FOR RULEMAKING

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Annette L. Vietti-Cook
Secretary
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Attention: Rulemakings and Adjudications Staff

PETITION FOR RULEMAKING

I. NEEDED REGULATION

This petition for rulemaking is submitted pursuant to 10 C.F.R. § 2.802 by Mark Edward Leyse (hereinafter "Petitioner").

Petitioner requests that the United States Nuclear Regulatory Commission ("NRC") require all holders of operating licenses for nuclear power plants ("NPP") to operate NPPs with in-core thermocouples[1] at different elevations and radial positions throughout the reactor core to enable NPP operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions. In the event of a severe accident, in-core thermocouples would enable NPP operators to accurately measure in-core temperatures, providing crucial information to help operators manage the accident; for example, indicating the time to transition from emergency operating procedures ("EOP") to implementing severe accident management guidelines ("SAMG").

II. STATEMENT OF PETITIONER'S INTEREST

On March 15, 2007, Petitioner submitted a petition for rulemaking, PRM-50-84 (ADAMS Accession No. ML070871368). PRM-50-84 was summarized briefly in American Nuclear Society's *Nuclear News*'s June 2007 issue[2] and commented on and deemed "a well-documented justification for...recommended changes to the [NRC's] regulations"[3] by Union of Concerned Scientists. In 2008, NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process.[4] And in 2009, NRC published

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[1] A thermocouple is a temperature-measuring device.
[2] American Nuclear Society, *Nuclear News*, June 2007, p. 64.
[3] David Lochbaum, Union of Concerned Scientists, "Comments on Petition for Rulemaking Submitted by Mark Edward Leyse (Docket No. PRM-50-84)," July 31, 2007, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML072130342, p. 2.
[4] Federal Register, Vol. 73, No. 228, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," November 25, 2008, pp. 71564-71569.

"Performance-Based Emergency Core Cooling System Acceptance Criteria," which gave advanced notice of a proposed rulemaking, addressing four objectives: the fourth being the issues raised in PRM-50-84.[5]

PRM-50-84 requests that NRC make new regulations: 1) to require licensees to operate LWRs under conditions that effectively limit the thickness of crud (corrosion products) and/or oxide layers on fuel cladding, in order to help ensure compliance with 10 C.F.R. § 50.46(b) ECCS acceptance criteria; and 2) to stipulate a maximum allowable percentage of hydrogen content in fuel cladding.

Additionally, PRM-50-84 requests that NRC amend Appendix K to Part 50—ECCS Evaluation Models I(A)(1), *The Initial Stored Energy in the Fuel*, to require that the steady-state temperature distribution and stored energy in the fuel at the onset of a postulated LOCA be calculated by factoring in the role that the thermal resistance of crud and/or oxide layers on cladding plays in increasing the stored energy in the fuel. PRM-50-84 also requested that these same requirements apply to any NRC-approved best-estimate ECCS evaluation models used in lieu of Appendix K to Part 50 calculations.

Petitioner also coauthored the paper, "Considering the Thermal Resistance of Crud in LOCA Analysis."[6]

III. BACKGROUND INFORMATION

A. The Need for Nuclear Power Plants to Operate with In-Core Thermocouples at Different Elevations and Radial Positions Throughout the Reactor Core

In October 1979, the President's Commission on the Three Mile Island accident recommended that:

> Equipment should be reviewed from the point of view of providing information to operators to help them prevent accidents and to cope with accidents when they occur. Included might be instruments that can provide proper warning and diagnostic information; for example, *the*

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[5] Federal Register, Vol. 74, No. 155, "Performance-Based Emergency Core Cooling System Acceptance Criteria," August 13, 2009, pp. 40765-40776.

[6] Rui Hu, Mujid S. Kazimi, Mark E. Leyse, "Considering the Thermal Resistance of Crud in LOCA Analysis," American Nuclear Society, 2009 Winter Meeting, Washington, D.C., November 15-19, 2009.

> *measurement of the full range of temperatures within the reactor vessel under normal and abnormal conditions*[7] [emphasis added].

In the last three decades, NRC has not made a regulation requiring that NPPs operate with in-core thermocouples at different elevations and radial positions throughout the reactor core to enable NPP operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions, which would help fulfill the President's Commission recommendations. If another severe accident were to occur in the United States, NPP operators would not know what the in-core temperatures were during the progression of the accident. In a severe accident, core-exit thermocouples would be the primary tool that was used to detect inadequate core cooling and core uncovery.

In a severe accident, in many cases, a predetermined core-exit temperature measurement (*e.g.*, 1200°F) would be used to signal the time for NPP operators to transition from EOPs to implementing SAMGs. For example, Westinghouse's probabilistic risk assessment for the AP1000 states:

> As the core-exit gas temperature increases above 1200 degrees [Fahrenheit], the EOPs transition to a red path indicating inadequate core cooling (FR-C.1). Upon entry into FR-C.1, the control room staff initiates actions to mitigate a severe accident by turning on the hydrogen igniters for hydrogen control and flooding the reactor cavity to prevent reactor pressure vessel failure.[8]

The problem with using a predetermined core-exit temperature measurement to signal the time for NPP operators to transition from EOPs to implementing SAMGs is that experimental data indicates that core-exit temperature ("CET") measurements have significant limitations: 1) "[t]he use of the CET measurements has limitations in detecting inadequate core cooling and core uncovery;" 2) "[t]he CET indication displays in all cases a significant delay (up to several 100 [seconds]);" and 3) "[t]he CET reading is

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[7] John G. Kemeny, *et al.*, "Report of the President's Commission on the Accident at Three Mile Island: The Need for Change: The Legacy of TMI," October 1979, p. 72.

[8] Westinghouse, "AP1000 Design Control Document," Rev. 19, Tier 2 Material, Chapter 19, "Probabilistic Risk Assessment," Appendix 19D, "Equipment Survivability Assessment," June 13, 2011, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML11171A416, p. 19D-3.

always significantly lower (up to several 100 [Kelvin]) than the actual maximum cladding temperature."[9]

Furthermore, in a severe accident experiment, the LOFT LP-FP-2 experiment, in which maximum fuel cladding temperatures exceeded 3308°F, the melting point of Zircaloy,[10] there was a time period that the measured CET was more than 2000°F lower than the maximum measured fuel cladding temperatures.[11] The substantial temperature differences of more than 2000°F between the measured CETs and maximum measured fuel cladding temperatures observed in LOFT LP-FP-2 indicate the magnitude that such temperature differences could be in an actual severe accident.

Unfortunately, despite the fact that "the nuclear industry developed SAMGs during the 1980s and 1990s in response to the [Three Mile Island] accident and followup activities," which "included extensive research and study (including several [probabilistic risk assessments]) on severe accidents and severe accident phenomena,"[12] NRC and the nuclear industry have ignored experimental data indicating that CET measurements have significant limitations. And ignored the President's Commission recommendations that NPPs have "instruments that can provide proper warning and diagnostic information; for example, the measurement of the full range of temperatures within the reactor vessel under normal and abnormal conditions."[13]

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[9] Robert Prior, *et al.*, OECD Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," NEA/CSNI/R(2010)9, November 26 2010, p. 128.

[10] NRC, "Feasibility Study of a Risk-Informed Alternative to 10 CFR 50.46, Appendix K, and GDC 35," June 2001, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML011800519, p. 3-1.

[11] Robert Prior, *et al.*, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," pp. 49-50.

[12] Charles Miller, *et al.*, NRC, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," SECY-11-0093, July 12, 2011, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML111861807, p. 47.

[13] John G. Kemeny, *et al.*, "Report of the President's Commission on the Accident at Three Mile Island: The Need for Change: The Legacy of TMI," p. 72.

B. NRC does Not Consider that Experimental Data Indicates that Core-Exit Temperature Measurements would Not be an Adequate Indicator for Detecting Inadequate Core Cooling and Core Uncovery in a Severe Accident

In July 2011, NRC's Near-Term Task Force, established in response to the Fukushima Dai-ichi accident, stated that "EOPs typically cover accidents to the point of loss of core cooling and initiation of inadequate core cooling (*e.g.*, core exit temperatures in PWRs greater than 649 degrees Celsius (1200 degrees Fahrenheit))."[14] In fact, Westinghouse's probabilistic risk assessment for the AP1000 states that in the event of a severe accident, as the CET exceeds 1200°F, "the control room staff initiates actions to mitigate a severe accident by turning on the hydrogen igniters for hydrogen control and flooding the reactor cavity to prevent reactor pressure vessel failure."[15]

Unfortunately, NRC and Westinghouse do not consider that experimental data from tests conducted at four facilities indicates that CET measurements would not be an adequate indicator for when to transition from EOPs to implementing SAMGs in a severe accident.[16]

Regarding 13 common conclusions made from the evaluation of tests conducted in four facilities (LOFT, PKL, ROSA/LSTF, and PSB-VVER) on CET measurements, an OECD Nuclear Energy Agency report, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," published in August 2010, states:

> 1) The use of the CET measurements has limitations in detecting inadequate core cooling and core uncovery.
>
> 2) The CET indication displays in all cases a significant delay (up to several 100 [seconds]).
>
> 3) The CET reading is always significantly lower (up to several 100 [Kelvin]) than the actual maximum cladding temperature.
>
> 4) CET performance strongly depends on the accident scenarios and the flow conditions in the core.

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[14] Charles Miller, *et al.*, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," p. 47.

[15] Westinghouse, "AP1000 Design Control Document," Rev. 19, Tier 2 Material, Chapter 19, "Probabilistic Risk Assessment," Appendix 19D, "Equipment Survivability Assessment," p. 19D-3.

[16] Robert Prior, *et al.*, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," pp. 128-129.

5) The CET reading depends on water fall-back from the upper plenum (due to; *e.g.*, reflux condensing [steam generator] mode or water injection) and radial core power profiles. During significant water fall-back the heat-up of the CET sensor could even be prevented.

6) The colder upper part of the core and the cold structures above the core are contributing to the temperature difference between the maximum temperature in the core and the CET reading.

7) The steam velocity through the bundle is a significant parameter affecting CET performance.

8) Low steam velocities during core boil-off are typical for [small-break loss-of-coolant accident] transients and can advance 3D flow effects.

9) In the core as well as above (*i.e.*, at the CET measurement level) a radial temperature profile is always measured (*e.g.*, due to radial core power distribution and additional effects of core barrel and heat losses).

10) Also at low pressure (*i.e.*, shut down conditions) pronounced delays and temperature differences are measured, which become more important with faster core uncovery and colder upper structures.

11) Despite the delay and the temperature difference the CET reading in the center reflects the cooling conditions in the core.

12) Any kind of [accident management] procedures using the CET indication should consider the time delay and the temperature difference of the CET behavior.

13) In due time after adequate core cooling is re-established in the core the CET reading corresponds to no more than the saturation temperature.[17]

(The LOFT facility was an actual nuclear reactor that was 1/50th the volume of a full-size PWR, "designed to represent the major component and system response of a commercial PWR."[18])

Regarding "two general limitations [that] have been identified regarding the ability of core exit fluid [thermocouples] to monitor a core uncovery"[19] in four tests

[17] *Id.*

[18] T. J. Haste, B. Adroguer, N. Aksan, C. M. Allison, S. Hagen, P. Hofmann, V. Noack, Organisation for Economic Co-Operation and Development, "Degraded Core Quench: A Status Report," August 1996, p. 13.

conducted in the LOFT facility, NUREG/CR-3386, "Detection of Inadequate Core Cooling with Core Exit Thermocouples: LOFT PWR Experience" published in November 1983, states:

> First, there was a delay between the core uncovery and the [thermocouple] response. This delay ranged from 28 to 182 [seconds] in the four LOFT LOCA simulations [discussed in this report], and could have been even longer in one case, had the reactor operators not initiated core reflood. The delay is judged to be caused by a film of water that coats the [thermocouple] and must be removed before the [thermocouple] can respond to the vapor superheat. The film of water exists due to slow drainage of liquid from the upper plenum. Although the magnitude of these delays is acceptable under the controlled conditions in the LOFT system, these delay times may differ in commercial systems and should be accounted for in the use of core exit [thermocouple] response to predict or measure [inadequate core cooling ("ICC")]. Since it is expected that ICC will initiate in the hottest core regions, any delay or inadequacy in measuring the temperature of these regions must be considered when analyzing potential methods for ICC detection.
>
> Second, the measured core exit [thermocouple] response was several hundred Kelvin lower than the maximum cladding temperatures in the core. This temperature difference results from the vapor superheat at the core exit being limited by the cladding temperatures near the core exit. In the LOFT system, these cladding temperatures were up to 360 K (648°F) lower than those in the high-power regions near the core center.
>
> In conclusion, any procedure that relies on the response of core exit fluid [thermocouples] to monitor a core uncovery should take these two limitations into account. There may be accident scenarios in which these [thermocouples] would not detect inadequate core cooling that preceded core damage.[20]

The four tests performed in the LOFT facility discussed in the quote above were the LOFT L2-5, L3-6/L8-1, L5-1, and L8-2 tests, which had maximum fuel cladding temperatures of 1479°F, 687°F, 828°F, and 1317°F, respectively.[21] The maximum fuel cladding temperatures in these four tests were more than 700°F below NRC's maximum

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[19] James P. Adams, Glenn E. McCreery, "Detection of Inadequate Core Cooling with Core Exit Thermocouples: LOFT PWR Experience," NUREG/CR-3386, EGG-2260, November 1983, p. 13.
[20] *Id.*
[21] *Id.*, p. 5.

fuel cladding temperature limit of 2200°F for design basis accidents.[22] Therefore, when measured CETs were several hundred degrees Fahrenheit lower—648°F in one case—than the maximum fuel cladding temperatures in the LOFT core, maximum fuel cladding temperatures were far below those of a severe accident.

In the severe accident temperature range—when maximum fuel cladding temperatures exceed 2200°F—it is probable that there would be far greater temperature differences between the measured CETs and maximum fuel cladding temperatures than was observed in the four LOFT facility tests discussed above, which simulated design basis accidents. In fact, significant temperature differences—greater than 2000°F—were observed in the final experiment conducted at the LOFT facility, LOFT LP-FP-2, a severe accident experiment, in which maximum fuel cladding temperatures exceeded 3308°F, the melting point of Zircaloy.

(LOFT LP-FP-2 is the only severe accident experiment that was an actual reactor core meltdown; it combined decay heating, severe fuel damage, and the quenching of Zircaloy cladding with water.[23] LOFT LP-FP-2 is considered "particularly important in that it was a large-scale integral experiment that provides a valuable link between the smaller-scale severe [accident] experiments and the TMI-2 accident."[24])

Regarding the significant temperature differences between measured CETs and maximum fuel cladding temperatures that were observed in LOFT LP-FP-2, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor" states:

> When the core temperatures started [thermal] runaway[25] at about 1500 [seconds after the experiment commenced] and quickly exceeded 2100 K

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[22] 10 C.F.R. § 50.46(b)(1)

[23] T. J. Haste, *et al.*, "Degraded Core Quench: A Status Report," p. 13.

[24] S. R. Kinnersly, *et al.*, "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," January 1991, p. 3.23.

[25] The initial heat up rate of the fuel cladding in LOFT LP-FP-2 was approximately 1.8°F per second. See T. J. Haste, *et al.*, "Degraded Core Quench: A Status Report," p. 13.

In LOFT LP-FP-2, at fuel cladding temperatures at which the zirconium-steam reaction became rapid, the local heat up rate of the fuel cladding began increasing. For example, at one location on the central fuel bundle (at the 42-inch elevation) when cladding temperatures had reached just below 2200°F, the fuel cladding heat up rate had increased to approximately 21.4°F per second; at the same location, between cladding temperatures of approximately 2200°F and 2780°F, the *average* heat up rate was approximately 36.3°F per second. See NRC, "Draft Interim Review of

> [3321°F] with a fission product release, the fluid temperatures in the upper plenum measured over the center fuel module...actually started to decrease. The temperature was typically 700 K [801°F] when quenching of the core occurred. For the peripheral bundles the temperatures were typically around 600 K [621°F] when core quench began.[26] ... The core quench caused a large excursion in the fluid temperature measurements. For a few seconds temperatures near 2000 K [3141°F] were observed followed by indication of saturation temperature.
>
> There was no evidence in the test that the CET indication was very much delayed. It can be concluded though that the core exit temperatures were much lower than typical core temperatures. During the rapid oxidation phase the CET appeared essentially to be disconnected from core temperatures. ... The temperature excursion at core quench is probably explained by a violent flow up through the bundle that heated up the thermocouples.[27]

In LOFT LP-FP-2, in a time period when maximum core temperatures were measured to exceed 3300°F, CETs were typically measured at 800°F—more than 2500°F lower than maximum core temperatures. And in LOFT LP-FP-2, "during the rapid oxidation phase the CET appeared essentially to be disconnected from core temperatures."[28]

The results of LOFT LP-FP-2 and other experiments demonstrate the need for NPPs to operate with in-core thermocouples at different elevations and radial positions throughout the reactor core to enable NPP operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions.

PRM-50-93/95 Issues Related to the LOFT LP-FP-2 Test," 2011, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML112650009, pp. 4, 5.

The phenomenon of rapid oxidation causing rapid fuel cladding temperature increases is sometimes termed "runway oxidation," "thermal runway," or "runway conditions." See Robert Prior, *et al.*, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," p. 130.

[26] The conductors of LOFT LP-FP-2 commenced reflooding the reactor core 1782.6 seconds after the experiment started. See J. P. Adams, *et al.*, "Quick Look Report on OECD LOFT Experiment LP-FP-2," OECD LOFT-T-3804, September 1985, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML071940358, Appendix E, p. E-17.

[27] Robert Prior, *et al.*, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," pp. 49-50.

[28] *Id.*, p. 50.

IV. THE RATIONAL FOR THE PROPOSED REGULATION

Petitioner is submitting this 10 C.F.R. § 2.802 petition because if NPPs were to operate with in-core thermocouples at different elevations and radial positions throughout the reactor core to enable NPP operators to accurately measure a large range of in-core temperatures in NPP steady-state and transient conditions, it would help improve public and plant-worker safety. In the event of a severe accident, in-core thermocouples would enable NPP operators to accurately measure in-core temperatures, providing crucial information to help operators manage the accident; for example, indicating the time to transition from EOPs to implementing SAMGs.

V. CONCLUSION

If implemented, the regulation proposed in this petition for rulemaking would help improve public and plant-worker safety.

Respectfully submitted,

/s/

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

Dated: February 28, 2012

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Tuesday, February 28, 2012 8:28 PM
To: Rulemaking Comments; Inverso, Tara; Dudley, Richard; Bladey, Cindy; PDR Resource
Cc: Christopher Paine; Thomas B. Cochran; Weaver, Jordan; Matthew G. McKinzie; Dave Lochbaum; Ed Lyman; Nuclear
Subject: Attention: Rulemakings and Adjudications Staff
Attachments: In-Core Thermocouples Rulemaking Petition.pdf

Dear Rulemaking and Adjudications Staff:

Attached to this e-mail is a 2.802 petition for rulemaking, dated February 28, 2012.
Sincerely,
Mark Leyse
P.O. Box 1314
New York, NY 10025
markleyse@gmail.com