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August 31, 2012

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

Attention: Rulemakings and Adjudications Staff

COMMENTS ON 10 CFR PARTS 50 AND 52; ONSITE EMERGENCY RESPONSE CAPABILITIES; NRC-2012-0031

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I. STATEMENT OF COMMENTER'S INTEREST

On March 15, 2007, Mark Edward Leyse, Commenter submitted a 10 C.F.R. § 2.802 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[1] and commented on and deemed "a well-documented justification for...recommended changes to the [United States Nuclear Regulatory Commission ("NRC")] regulations"[2] by Union of Concerned Scientists. In 2008, NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process.[3] And in 2009, NRC published "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.[4]

PRM-50-84 requests that NRC make new regulations: 1) to require licensees to operate light water reactors 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) emergency core cooling systems ("ECCS") acceptance criteria; and 2) to stipulate a maximum allowable percentage of hydrogen content in fuel cladding.

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[1] American Nuclear Society, *Nuclear News*, June 2007, p. 64.
[2] 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.
[3] Federal Register, Vol. 73, No. 228, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," November 25, 2008, pp. 71564-71569.
[4] Federal Register, Vol. 74, No. 155, "Performance-Based Emergency Core Cooling System Acceptance Criteria," August 13, 2009, pp. 40765-40776.

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 loss-of-coolant accident ("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.

Commenter also coauthored the paper, "Considering the Thermal Resistance of Crud in LOCA Analysis."[5]

On February 28, 2012, Commenter submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-105 (ADAMS Accession No. ML12065A215), to request that NRC require all holders of operating licenses for nuclear power plants ("NPP") to operate NPPs with in-core thermocouples installed 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 track the progression of core damage and manage the accident; for example, indicating the time to transition from emergency operating procedures ("EOP") to implementing severe accident management guidelines ("SAMG").

II. COMMENTS ON ADVANCE NOTICE OF THE PROPOSED RULEMAKING

In NRC's April 2012 Federal Register notice for a proposed rulemaking, regarding onsite emergency response capabilities (questions of Section IV.B.2.b.), NRC asks stakeholders:

> What is the best approach to ensure that procedural guidance for beyond design basis events is based on sound science, coherent, and integrated? What is the most effective strategy for linking the EOPs with the SAMGs and [Extreme Damage Mitigation Guidelines ("EDMG")]? Should the

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

> transition from EOPs to SAMGs be based on key safety functions, or should the SAMGs be developed in a manner that addresses a series of events that are beyond a plant's design basis?[6]

In these comments, Commenter provides information pertinent to the questions of Section IV.B.2.b. of NRC's April 2012 Federal Register notice: Commenter provides information drawing attention to problems with relying on core exit thermocouple ("CET") readings in a severe accident. For pressurized water reactor ("PWR") severe accidents, CET readings would be used to signal the point for plant operators to transition from EOPs to implementing SAMGs (discussed in Section II.A of these comments).

(Boiling water reactors ("BWR") do not have CETs.)

NRC and the U.S. nuclear industry need to acknowledge that there is experimental data from tests conducted at four facilities indicating there would be problems with relying on CET readings in the event of a severe accident. This experimental data is discussed in a 2010 OECD Nuclear Energy Agency report, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor"[7] (and discussed in Section II.B of these comments).

On February 28, 2012, Commenter submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-105, proposing an alternative to relying on CET readings in the event of a severe accident (discussed in Section II.C of these comments).

PRM-50-105 requests that NRC require that NPPs operate with in-core thermocouples installed 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 track the progression of core damage and manage the accident; for example, indicating the time to transition from EOPs to implementing SAMGs.

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[6] NRC, "Onsite Emergency Response Capabilities: Advance Notice of Proposed Rulemaking," 10 CFR Parts 50 and 52, NRC–2012–0031, Federal Register, Vol. 77, No. 75, April 18, 2012, p. 23164.

[7] 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.

(Plant operators at both PWRs and BWRs would benefit from relying on in-core thermocouple readings in the event of a severe accident.)

A. In a PWR Severe Accident, Core Exit Temperature Measurements Would Be Used to Signal the Point to Transition from EOPs to SAMGs

In Nuclear Energy Institute's ("NEI") comments on NRC's April 2012 Federal Register notice for a proposed rulemaking, regarding onsite emergency response capabilities, NEI explains that for the nuclear industry, "[Electric Power Research Institute ("EPRI")] has responsibility for developing the [1992] Technical Basis Report (TBR),[8] which provides the fundamental scientific understanding of severe accident phenomenology and its translation into 'candidate high-level actions' to mitigate the accident progression."[9] NEI also explains that the reactor vendor owners groups use EPRI's TBR to develop SAMGs for the different NPP designs licensed by NRC, which consider the different types of reactors and containments.[10]

In NRC's April 2012 notice for the proposed rulemaking (questions of Section IV.B.2.b.), NRC asks stakeholders:

> What is the best approach to ensure that procedural guidance for beyond design basis events is based on sound science, coherent, and integrated? What is the most effective strategy for linking the EOPs with the SAMGs and [Extreme Damage Mitigation Guidelines ("EDMG")]? Should the transition from EOPs to SAMGs be based on key safety functions, or should the SAMGs be developed in a manner that addresses a series of events that are beyond a plant's design basis?[11]

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[8] EPRI, "Severe Accident Management Guidance Technical Basis Report," EPRI TR-101869, December 1992, Volume 1: "Candidate High-Level Actions and Their Effects" and Volume 2: "The Physics of Accident Progression."

[9] Anthony R. Pietrangelo, Senior Vice President and Chief Nuclear Officer, NEI, "Comments on Advance Notice of Proposed Rulemaking; Docket ID NRC-2012-031; 10 CFR Part 50 and 52, Onsite Emergency Response Capabilities (77 Fed. Reg. 23161)," June 18, 2012, p. 2.

[10] *Id.*

[11] NRC, "Onsite Emergency Response Capabilities: Advance Notice of Proposed Rulemaking," 10 CFR Parts 50 and 52, NRC–2012–0031, Federal Register, Vol. 77, No. 75, April 18, 2012, p. 23164.

NEI responded to NRC's questions. Among other statements in NEI's response, NEI answered:

> [T]he EPRI TBR, in conjunction with many other source documents such as NUREGs, provides a sound scientific foundation, and this is supplemented in application by insights from the plant [probabilistic risk assessment (PRA)] and other plant-specific information. There are currently well defined transitions from the EOPs, which are focused on preventing core damage, to the SAMGs, which are focused on protecting fission-product boundaries once it is determined that core damage cannot be prevented. ...
>
> A transition from an EOP to a SAMG should be symptom-based; *i.e.*, based upon control room receipt of specific parameter values that directly indicate incipient core damage. The transition is clear and is reinforced through training. This symptom-based approach, which is independent of the initiating event, is currently used by the industry and should be maintained.[12]

Indeed, for PWR severe accidents, there is a "well-defined transition" from EOPs to SAMGs. For PWRs, which have core exit thermocouples ("CET"), in a severe accident, plant operators are instructed to transition from EOPs to SAMGs when CET readings exceed 1200°F. (The same "well-defined transition" from EOPs to SAMGs would also be used for the AP1000 design.[13])

In Westinghouse's probabilistic risk assessment for the AP1000, Westinghouse defines two of the time frames that would occur in a severe accident: Time Frame 1 is the Core Heatup Phase and Time Frame 2 is the In-Vessel Severe Accident Phase. Westinghouse states that "Time Frame 1 is defined as the period of time after core uncovery and prior to the onset of significant core damage as evidenced by the rapid zirconium-water reactions in the core. This is the transition period from design basis to severe accident environment."[14] Regarding Time Frame 2, Westinghouse states that "[t]he onset of rapid zirconium-water reactions of the fuel rod cladding and hydrogen

[12] Anthony R. Pietrangelo, "Comments on Advance Notice of Proposed Rulemaking; Docket ID NRC-2012-031; 10 CFR Part 50 and 52, Onsite Emergency Response Capabilities (77 Fed. Reg. 23161)," Attachment 1, p. 3.

[13] 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.

[14] *Id.*

generation defines the beginning of Time Frame 2. The heat of the exothermic reaction accelerates the degradation, melting, and relocation of the core."[15]

Westinghouse's probabilistic risk assessment for the AP1000 states that the core-exit gas temperature would reach 1200°F in Time Frame 1, before the onset of the rapid zirconium-steam reaction of the fuel cladding.[16]

In a different Westinghouse document, from 2008, Westinghouse states that "[a]n inadequate core cooling condition is assumed in the [Westinghouse Owners Group[17] emergency response guidelines] if the highest reading CETs are indicating greater than 1200 degrees F."[18] Therefore, according to Westinghouse, CET readings of 1200°F are a primary symptom of an inadequate core cooling condition.

The U.S. nuclear industry and NRC both assume that CET readings of 1200°F are a primary symptom of an inadequate core cooling condition. (For example, 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))."[19])

As instrumentation, CETs are required to have the highest standards for functioning in the event of an accident.[20] (CETs are required to be Category 1 instrumentation[21] (a non-legally binding requirement). NRC states that "[i]n general,

[15] *Id.*, p. 19D-3, 19D-4.
[16] *Id.*, p. 19D-3.
[17] The Westinghouse Owners Group ("WOG") is now named the Pressurized Water Reactor Owners Group ("PWROG").
[18] Robert J. Lutz, Jr., Westinghouse Electric Company, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," WCAP-15981-NP-A, Revision 0, September 2008, located at: www.nrc.gov, Electronic Reading Room, ADAMS Documents, Accession Number: ML103560687, p. 60.
[19] 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.
[20] Robert J. Lutz, Jr., "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," WCAP-15981-NP-A, p. xxii.
[21] *Id.*

Category 1 provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power," which would include seismic qualification.[22])

Westinghouse certainly believes that CETs are essential instrumentation. In a 2008 document, Westinghouse states:

> The new core damage assessment methodology relies solely on instrumentation to determine the occurrence of and degree of core damage. The methodology uses two primary indicators, based on the analytical modeling of a wide range of core damage accidents: 1) CETs and 2) containment radiation.[23]

And in the same 2008 document, Westinghouse states:

> [T]he CET indication provides the most direct and unambiguous indication of the potential loss of fuel rod clad barrier. ...
>
> The loss of fuel rod clad barrier will always be indicated first by high CET indications. Containment and [reactor coolant system] letdown radiation levels will always lag the CET temperatures and may be useful only to confirm the loss of the fuel rod clad barrier. The issue with the radiation monitors is that a pathway must exist for the fission products to reach the volume being monitored for high radiation levels.[24]

In fact, Westinghouse has "concluded that only the [CETs] can provide a direct indication of core cooling."[25] However, the experimental data discussed in Section II.B of these comments indicates that CET readings could be inadequate for providing information to help plant operators initiate crucial accident management actions.

B. Experimental Data Demonstrates Problems with Using Core Exit Temperature Measurements to Signal the Point to Transition from EOPs to SAMGs

In 2005, an experiment simulating a small beak LOCA (test 6-1) was conducted in Japan at the Large Scale Test Facility ("LSTF") as part of the OECD ROSA/LSTF

[22] NRC, Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML003740282, p. 1.97-4.

[23] Robert J. Lutz, Jr., "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," WCAP-15981-NP-A, p. 29.

[24] *Id.*, p. 26.

[25] *Id.*, Appendix B, "PWROGs Responses to the NRCs Requests for Additional Information," p. B-23.

project; Test 6-1 "had to be terminated prematurely to avoid excessive overheating of the core." A 2010 OECD Nuclear Energy Agency report, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," states that "[i]t was noted that the test 6-1 results showed that the core uncovery had started significantly early before the [CETs] indicated superheating and that the temperature increase rate was higher in the core than in the CETs. The results suggested that the response of the [CETs] could be inadequate to initiate the relevant [accident management] actions."[26]

The 2010 OECD Nuclear Energy Agency report also states that experimental data regarding how CETs performed in other tests—conducted in the LOFT, PKL, ROSA/LSTF, and PSB-VVER facilities—confirmed that CET readings could be inadequate for providing information to help plant operators initiate crucial accident management actions.[27]

Additionally, the 2010 OECD Nuclear Energy Agency report states that "[CETs] are generally installed *much closer* to the core in the experiments than in [NPPs]—in LOFT, only one inch above the core. Therefore it may *not* be assumed that the CETs in [NPPs would] behave more favorably than in the experiments"[28] [emphasis added].

NRC and the U.S. nuclear industry need to acknowledge that there is experimental data from tests conducted at four facilities indicating that CET measurements would not be an adequate indicator for when to transition from EOPs to implementing SAMGs in a severe accident.[29]

Regarding 13 common conclusions made from the evaluation of tests conducted in four facilities (LOFT, PKL, ROSA/LSTF, and PSB-VVER) on CET measurements, the 2010 OECD Nuclear Energy Agency report 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]).

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[26] Robert Prior, *et al.*, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," NEA/CSNI/R(2010)9, p. 11.
[27] *Id.*
[28] *Id.*
[29] *Id.*, pp. 128-129.

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.
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.[30]

(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."[31])

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[30] *Id.*

Regarding "two general limitations [that] have been identified regarding the ability of core exit fluid [thermocouples] to monitor a core uncovery"[32] in four tests 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.[33]

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

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[31] 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.

[32] 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.

[33] *Id.*

temperatures of 1479°F, 687°F, 828°F, and 1317°F, respectively.[34] The maximum fuel cladding temperatures in these four tests were more than 700°F below NRC's maximum fuel cladding temperature limit of 2200°F for design basis accidents.[35] Therefore, when measured CETs were several hundred 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, the LOFT LP-FP-2 experiment, a severe accident experiment, in which maximum fuel cladding temperatures exceeded 3308°F, the melting point of Zircaloy.[36]

(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.[37] 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."[38])

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[34] *Id.*, p. 5.
[35] 10 C.F.R. § 50.46(b)(1)
[36] 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.
[37] T. J. Haste, *et al.*, "Degraded Core Quench: A Status Report," p. 13.
[38] 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.

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[39] at about 1500 [seconds after the experiment commenced] and quickly exceeded 2100 K [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.[40] ... 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.[41]

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[39] 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 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.

[40] 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.

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

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."[42]

The results of LOFT LP-FP-2 and other experiments demonstrate problems with using CET measurements to signal the point to transition from EOPs to implementing SAMGs. EPRI's 1992 Technical Basis Report[43]—which, according to NEI, "provides the fundamental scientific understanding of severe accident phenomenology" for the U.S. nuclear industry[44]—is simply incorrect regarding the effectiveness of CETs for tracking the progression of core damage in a severe accident.

C. In-Core Thermocouples Would Be Superior to Core Exit Thermocouples for Signaling the Point to Transition from EOPs to SAMGs

In NRC's April 2012 notice for the proposed rulemaking (questions of Section IV.B.2.b), NRC asks stakeholders:

> What is the best approach to ensure that procedural guidance for beyond design basis events is based on sound science, coherent, and integrated? What is the most effective strategy for linking the EOPs with the SAMGs and [Extreme Damage Mitigation Guidelines ("EDMG")]? Should the transition from EOPs to SAMGs be based on key safety functions, or should the SAMGs be developed in a manner that addresses a series of events that are beyond a plant's design basis?[45]

In a PWR severe accident, CET readings would be used to signal the point for plant operators to transition from EOPs to implementing SAMGs (discussed in Section II.A of these comments). However, there is experimental data from tests conducted at

[42] *Id.*, p. 50.
[43] EPRI, "Severe Accident Management Guidance Technical Basis Report," EPRI TR-101869, December 1992, Volume 1: "Candidate High-Level Actions and Their Effects" and Volume 2: "The Physics of Accident Progression."
[44] Anthony R. Pietrangelo, "Comments on Advance Notice of Proposed Rulemaking; Docket ID NRC-2012-031; 10 CFR Part 50 and 52, Onsite Emergency Response Capabilities (77 Fed. Reg. 23161)," p. 2.
[45] NRC, "Onsite Emergency Response Capabilities: Advance Notice of Proposed Rulemaking," 10 CFR Parts 50 and 52, NRC–2012–0031, Federal Register, Vol. 77, No. 75, April 18, 2012, p. 23164.

four facilities indicating there would be problems with relying on CET readings in the event of a severe accident (discussed in Section II.B of these comments).

On February 28, 2012, Commenter submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-105, proposing an alternative to relying on CET readings in the event of a severe accident. PRM-50-105 requests that NRC require that NPPs operate with in-core thermocouples installed 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.

Plant operators at both PWRs and BWRs would benefit from relying on in-core thermocouple readings in the event of a severe accident. In such an accident, in-core thermocouples would provide NPP operators with crucial information to help them track the progression of core damage and manage the accident; for example, indicating the correct time to transition from EOPs to implementing SAMGs.

It is clear from the experimental data discussed in Section II.B of these comments that in-core thermocouple measurements would be superior to core exit thermocouple measurements for diagnosing the condition of a reactor core in the event of a severe accident. After all, in the experiments discussed in Section II.B, it was the in-core thermocouple measurements which revealed how much lower the core exit temperatures were than the in-core temperatures.

1. In-Core Thermocouples have Been Tested and Used in Nuclear Reactors for Decades

In-core thermocouples have been tested and used in nuclear reactors for decades, as the primary component of in-core gamma thermometers, which are "device[s] used for measuring the gamma flux in a nuclear reactor."[46]

"Instrumentation and Control Systems," Chapter 7 of "ESBWR Design Control Document," states that gamma thermometers—the present Radcal design[47]—have been

[46] GE Hitachi Nuclear Energy, "Licensing Topical Report: Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," NEDO-33197-A, Revision 3, Class I, October 2010, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML102810320, p. 1.

installed in various nuclear reactors since 1979.[48] For example, Radcal gamma thermometers were installed in PWRs in Palisades Nuclear Plant and Arkansas Nuclear One Units 1 and 2 in the 1980s.

Radcal gamma thermometers have also been installed in BWRs. GE Hitachi Nuclear Energy, "Licensing Topical Report: Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," states that "[t]here have been three in-plant tests of [gamma thermometer] sensors in BWRs thus far. The first test was at Limerick 2 and lasted for two cycles, a total of four years. The second test, which was at Tokai 2, lasted for a single cycle of one year duration."[49] The third test was conducted at Kashiwazaki-Kariwa 5.[50]

GE Hitachi Nuclear Energy certainly seems to be satisfied with the in-core performance of Radcal gamma thermometers—which each have two thermocouples—because GE Hitachi Nuclear Energy has plans to install Radcal gamma thermometers in the Economic Simplified Boiling Water Reactor ("ESBWR").

(The two thermocouples within each gamma thermometer are internally connected, which enables them to measure the temperature difference between two separate points that are located within the gamma thermometer—only the temperature difference is measured for plant operators.)

(It is noteworthy that a 2009 Idaho National Laboratory ("INL") report, "High Temperature Irradiation-Resistant Thermocouple Performance Improvements," states that INL has "developed and evaluated the performance of a high temperature irradiation-resistant thermocouple...that contains doped molybdenum and a niobium alloy. Data from high temperature (up to 1500°C), long duration (up to 4000 hours) tests and on-going irradiations at INL's Advanced Test Reactor demonstrate the superiority of these sensors to commercially-available thermocouples. However, several options have been

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[47] R. H. Leyse, R. D. Smith: "Gamma Thermometer Developments for Light Water Reactors," IEEE Transactions on Nuclear Science, Vol.N5.26, No. 1, February 1979, pp. 934–943.
[48] GE Nuclear Energy, "ESBWR Design Control Document," Tier 2, Chapter 7, "Instrumentation and Control Systems," 26A6642AW, Revision 1, January 2006, available at: www.nrc.gov, NRC Library, ADAMS Documents, Accession Number: ML060520260, pp. 7A-6, 7A-7.
[49] GE Hitachi Nuclear Energy, "Licensing Topical Report: Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," NEDO-33197-A, p. 25.
[50] *Id.*

identified that could further enhance their reliability, *reduce their production costs, and allow their use in a wider range of operating conditions*" [emphasis added].[51]

The INL report also states that high temperature irradiation-resistant thermocouples can be developed for specific customer needs and varied conditions.[52])

2. GE Hitachi Nuclear Energy has Plans to Install In-Core Thermocouples in the ESBWR

GE Hitachi Nuclear Energy's licensing topical report, "Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," states that there are plans to install instrument tubes, which each have seven gamma thermometers at different elevations, at 64 radial positions throughout the reactor core of the ESBWR.[53] Thermocouples would be the primary component of the gamma thermometers installed in the reactor core of the ESBWR.

Each gamma thermometer has two thermocouples; therefore, GE Hitachi Nuclear Energy has plans to install 896 in-core thermocouples in each ESBWR reactor.

3. According to GE Hitachi Nuclear Energy Maintaining In-Core Thermocouples Would Cause Virtually No Radiation Dose to Workers

According to GE Hitachi Nuclear Energy, "A [gamma thermometer] system...has no moving parts, no under vessel tubing, *virtually no radiation dose* to maintenance since it is a fixed in-core probe, and is expected to be very reliable"[54] [emphasis added]. In-core thermocouples could certainly be placed inside of instrument tubes, distributed throughout the reactor core, like gamma thermometers are; hence, according to GE Hitachi Nuclear Energy, in-core thermocouples would cause virtually no radiation dose to workers during maintenance.

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[51] Joshua Daw, *et al.*, Idaho National Laboratory, "High Temperature Irradiation-Resistant Thermocouple Performance Improvements," INL/CON-09-15267, Sixth American Nuclear Society International Topical Meeting on Nuclear Plant Instrumentation, Control, and Human-Machine Interface Technologies, April 2009, p 1.
[52] *Id.*
[53] GE Hitachi Nuclear Energy, "Licensing Topical Report: Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring," NEDO-33197-A, p. 8.
[54] *Id.*, p. 1.

(It also follows that GE Hitachi Nuclear Energy claims that in-core thermocouples would be very reliable, because thermocouples are the primary component of gamma thermometers.)

III. CONCLUSION

Plant operators at both PWRs and BWRs would benefit from relying on in-core thermocouple readings in the event of a severe accident. In such an accident, in-core thermocouples would provide NPP operators with crucial information to help them track the progression of core damage and manage the accident; for example, indicating the correct time to transition from EOPs to implementing SAMGs.

If in-core thermocouples were installed at different elevations and radial positions throughout NPP reactor cores, it would help improve public and plant-worker safety.

Respectfully submitted,

/s/

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Dated: August 31, 2012

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Friday, August 31, 2012 3:43 PM
To: Rulemaking Comments
Cc: PDR Resource; Bladey, Cindy; Inverso, Tara
Subject: NRC-2012-0031
Attachments: Comments on NRC Rulemaking; NRC-2012-0031.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail are Mark Leyse's comments, dated August 31, 2012, regarding NRC's notice of solicitation of public comments on a proposed rulemaking, regarding onsite emergency response capabilities, 10 CFR Parts 50 and 52, NRC-2012-0031, published in the Federal Register on April 18, 2012.

Sincerely,

Mark Leyse