PRM-50-105
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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

August 6, 2012

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

Attention: Rulemakings and Adjudications Staff

COMMENTS ON PRM-50-105; NRC-2012-0056

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I. 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[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 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.

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

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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.

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.

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

Petitioner has submitted PRM-50-105 to request that NRC require all holders of operating licenses for nuclear power plants ("NPP") to operate NPPs 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. 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. SUPPLEMENTARY INFORMATION ON PRM-50-105

A. In-Core Thermocouples Would Satisfy the Near-Term Task Force Report on Insights from the Fukushima Dai-ichi Accident Recommendations for Enhanced Reactor Instrumentation

An April 2012 Advisory Committee on Reactor Safeguards ("ACRS") report states that "NRC has recognized the need for enhanced reactor...instrumentation and is in the process of adding this to the implementation of the [Near-Term Task Force report on insights from the Fukushima Dai-ichi accident] recommendations."[6] And the Near-Term Task Force report "recommends strengthening and integrating onsite emergency response

[5] 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.

[6] ACRS, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program: A Report to the U.S. Nuclear Regulatory Commission," NUREG-1635, Vol. 10, April 2012, p. 12.

capabilities such as EOPs [and] SAMGs;"[7] the April 2012 ACRS report states that "[s]uch integration could focus on the need to clarify the transition points"[8] that would occur in a NPP accident.

In-core thermocouples—which would measure a wide range of temperatures inside the reactor core under typical and accident conditions—would fulfill the need for enhanced reactor instrumentation. In-core thermocouples would provide NPP operators with crucial information to help them track the progression of core damage and manage an accident; for example, indicating the correct time to transition from EOPs to implementing SAMGs.

B. Boiling Water Reactors Need to Operate with In-Core Thermocouples

Core-exit thermocouples are not installed in boiling water reactors ("BWR").[9] In the event of a severe accident, BWR plant operators are supposed to detect inadequate core cooling and core uncovery by measuring the water level in the reactor core. However, "BWR high drywell temperature and low pressure accidents ([for example], LOCAs) can cause the water level to read erroneously high...and BWR water level readings are unreliable after core damage."[10]

In a December 2011 article, Saloman Levy stated that in the Fukushima Dai-ichi accident, plant operators should have "[r]ecognize[d] unreliable water level data and avoid[ed] their use to delay urgent actions."[11] The same article states that "[t]he reactor and containment pressures and the wetwell water temperature would be superior alternate indicators [than water level measurements]. ... The reactor and the containment

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[7] 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, pp. ix, 49, 69.

[8] ACRS, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program: A Report to the U.S. Nuclear Regulatory Commission," NUREG-1635, p. 11.

[9] IAEA, "Generic Assessment Procedures for Determining Protective Actions during a Reactor Accident," IAEA-TECDOC-955, August 1997, p. 25.

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

[11] Saloman Levy, "How Would U.S. Units Fare?," Nuclear Engineering International, December 7, 2011. Levy makes a point of qualifying that his observations are not intended to be criticisms of the actions of the Fukushima Dai-ichi plant operators.

pressures will rise faster when hydrogen is produced. Increased reactor and containment pressure rates and wetwell [water] temperature rises confirm accelerated core melt."[12]

In his article, Levy concludes "that formation of hydrogen and the acceleration in the rate of its formation need to be forecasted and detected to shift top priority to reactor water addition and to assure its success."[13] The problem with what Levy suggests is simply that by the time operators *confirmed an accelerated core melt*—by measuring increased reactor and containment pressure rates and/or wetwell water temperature rises—the reactor core would already be overheated and reflooding an overheated core could generate hydrogen, at rates as high as 5.0 kg per second.[14]

It is clear that in the event of a BWR severe accident, in-core thermocouple measurements would be more accurate and immediate for detecting inadequate core cooling and core uncovery than readings of the reactor water level, reactor pressure, containment pressure, or wetwell water temperature.

C. Problems with the Use of Core Exit Thermocouples in Westinghouse's Emergency Response Guidelines 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."[15]

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[12] Saloman Levy, "How Would U.S. Units Fare?," Nuclear Engineering International, December 7, 2011.
[13] *Id.*
[14] E. Bachellerie, *et al.*, "Generic Approach for Designing and Implementing a Passive Autocatalytic Recombiner PAR-System in Nuclear Power Plant Containments," Nuclear Engineering and Design, 221, 2003, p. 158.
[15] 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.

Regarding Time Frame 2, Westinghouse states that "[t]he onset of rapid zirconium-water reactions of the fuel rod cladding and hydrogen generation defines the beginning of Time Frame 2. The heat of the exothermic reaction accelerates the degradation, melting, and relocation of the core."[16]

Westinghouse maintains 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.[17] However, experimental data demonstrates that this would not necessarily be the case.

In the LOFT LP-FP-2 experiment, an experiment simulating a severe accident, core-exit temperatures were measured at around 800°F when in-core thermocouples measured fuel cladding temperatures exceeding 3300°F. Therefore, after the onset of the rapid zirconium-steam reaction, core-exit temperatures were measured at around 800°F.

An OECD Nuclear Energy Agency report, "Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor," published in 2010, states that in LOFT LP-FP-2, "during the rapid oxidation phase [core-exit temperatures] appeared essentially to be disconnected from core temperatures."[18]

Clearly, there are problems with Westinghouse's emergency response guidelines for the AP1000. Plant operators are instructed to actuate the AP1000 containment hydrogen igniters after the core-exit thermocouple measurements exceed 1200°F, which would most likely be some time after a meltdown had commenced.

Another problem with Westinghouse's plan to have plant operators rely on core-exit thermocouple measurements in the event of a severe accident is that plant operators might reflood an overheated core when they did not realize that the core was in fact overheated. Consider a scenario in which there were similar temperature differences between in-core and core-exit temperatures as were observed in LOFT LP-FP-2. If plant operators were to reflood the core when core-exit temperatures were well below 1200°F, the core could already be overheated—fuel-cladding temperatures could be over 3300°F,

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[16] *Id.*, p. 19D-3, 19D-4.
[17] *Id.*, p. 19D-3.
[18] 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. 50.

at a temperature where zirconium melts. In such a case, there would also be some liquefaction of core components because of eutectic reactions taking place at temperatures as low as 2200°F. For example, the eutectic reaction between zirconium and stainless steel.

Unintentionally reflooding an overheated core could be very dangerous. In a severe accident, during the reflooding of an overheated reactor core up to 300 kilograms of hydrogen could be generated in one minute.[19]

Regarding the refooding of an overheated reactor core, a second OECD Nuclear Energy Agency report, "In-Vessel Core Degradation Code Validation Matrix: Update 1996-1999," published in 2000, states:

> Several of the integrated core damage progression tests have been reflooded, resulting in production of significant amounts of steam, with further oxidation and hydrogen generation, as observed in some CORA tests and in LOFT-LP-FP-2. This renewed heatup is important regarding accident management, as the additional hydrogen might threaten containment integrity and increased fission product release would increase the source term. The increasing fuel temperatures, being counter-intuitive, might confuse the operators into taking inappropriate action.[20]

It is evident that with Westinghouse's plan to have plant operators rely on core-exit thermocouple measurements in the event of a severe accident, operators could unintentionally reflood an overheated core, which would rapidly generate additional hydrogen, at a rate as high as 5.0 kilograms per second, which could, in turn, compromise the containment if the hydrogen were to detonate.

Two of the main conclusions from data from experiments simulating design basis accidents conducted at four different facilities are that core exit temperature measurements display in all cases a significant delay (up to several hundred seconds) and that core exit temperature measurements are always significantly lower (up to several hundred Celsius) than the actual maximum cladding temperature.

Clearly, for severe accidents, Westinghouse's plan for AP1000 plant operators to rely on core exit temperature measurements to monitor the condition of the core and to

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[19] E. Bachellerie, *et al.*, "Generic Approach for Designing and Implementing a Passive Autocatalytic Recombiner PAR-System in Nuclear Power Plant Containments," p. 158.
[20] OECD Nuclear Energy Agency, "In-Vessel Core Degradation Code Validation Matrix: Update 1996-1999," Report by an OECD NEA Group of Experts, October 2000, p. 13.

wait for a core exit temperature measurement of 1200°F to signal when to actuate hydrogen igniters and implement other procedures would be neither productive nor safe.

III. CONCLUSION

If implemented, the regulations proposed in PRM-50-105 would help improve public and plant-worker safety.

Respectfully submitted,

/s/

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Dated: August 6, 2012

Rulemaking Comments

From: Mark Leyse [markleyse@gmail.com]
Sent: Sunday, August 05, 2012 5:14 PM
To: Rulemaking Comments
Cc: PDR Resource; Bladey, Cindy
Subject: NRC-2012-0056
Attachments: Comments on PRM-50-105.pdf

Dear Ms. Vietti-Cook:

Attached to this e-mail is Mark Leyse's, Petitioner's, response, dated August 6, 2012, to the NRC's notice of solicitation of public comments on PRM-50-105, NRC-2012-0056, published in the Federal Register on May 23, 2012.

Sincerely,

Mark Leyse