NRR-PMDAPEm Resource

From: Dave Lochbaum [DLochbaum@ucsusa.org]
Sent: Friday, May 08, 2015 12:00 PM
To: Lund, Louise; Lamb, John
Subject: Comments on proposed director's decision - BWR Mark I and II containment petition
Attachments: 20150508-ucs-nrc-gdc-44-petition-directors-decision-comments.pdf

Hello Louise and John:

Attached please find a digital copy with our comments on the proposed director's decision.

I don't plan on also mailing in a hard copy of the letter, but would do so upon request.

Thanks,
Dave Lochbaum
UCS

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From: Dave Lochbaum

Created By: DLochbaum@ucsusa.org

Recipients:
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Tracking Status: None
"Lamb, John" <John.Lamb@nrc.gov>
Tracking Status: None

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May 8, 2015

Ms. Luise Lund, Acting Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Comments on Proposed Director's Decision for 10 CFR 2.206 Petition Seeking Demands For Information to Boiling Water Reactors with Mark I and II Containment Designs

Dear Ms. Lund:

In response to your letter dated April 17, 2015, I am providing comments on the proposed director's decision for the petition submitted July 29, 2011. I appreciate the implied compliment the NRC paid me in thinking I would be able to review and comment on the 24-page proposed director's decision within 30 days, considering it took the NRC staff 1,358 days to review and comment on my 8-page petition. I am pleased to meet the NRC's expectations by submitting comments several under the 30-day deadline.

As stated in the attached comments, UCS believes the Demands For Information are still necessary to provide reasonable assurance that the regulatory requirements in General Design Criterion 44 and 10 CFR 50.49 are being met.

Sincerely,

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Enclosure: Comments on Proposed Director's Decision

Section	Comment
Page 5, first full paragraph and page 8, paragraph before the Conformance with the Environmental Qualification Regulation section	The Proposed Director's Decision states that the NRC considers General Design Criterion 61 to be applicable to the spent fuel pool heat removal function and that the spent fuel pool cooling system need not comply with General Design Criterion 44. **UCS Comment:** It is clear and unequivocal that nuclear power reactors licensed by the NRC must comply with all applicable regulatory requirements, including GDC 44 and GDC 61. UCS agrees with the NRC staff that spent fuel pools and their cooling and makeup systems must comply with GDC 61. UCS also agrees with the NRC staff that containments and their cooling systems must comply with GDC 44. Because the spent fuel pools at boiling water reactors with Mark I and II containment designs – the subjects of our petition – are located inside containment, compliance with GDC 44 inherently includes handling the heat released from the spent fuel pools and the cooling/makeup equipment operating to support them. In other words, one cannot pretend that these heat loads do not exist and still be in compliance with GDC 44. There is an overlap between GDC 61 and GDC 44 forced by the location of the spent fuel pools and support equipment within containment. Similar overlaps exist between GDC 33 and GDC 44 as well as between GDC 35 and GDC 44 and the NRC requires compliance with GDC 33 GDC 35 as well as with GDC 44. The NRC must treat GDC 44 and GDC 61 the same way. GDC 33, Reactor Coolant Makeup, requires a system to provide makeup to the reactor vessel in event of small breaks in the reactor coolant system. The High Pressure Coolant Injection (HPCI) system at most BWRs fulfills this requirement. Because the HPCI steam turbine, pump, and piping are located within secondary containment, GDC 44 calculations and evaluations must account for the heat loads from HPCI operation during an accident. If the heat loads from HPCI operation during an accident had been excluded from GDC calculations, it would be immaterial with respect to GDC 44 compliance whether the HPCI design complied with GDC 33. Both GDC must be met. GDC 35, Emergency Core Cooling, requires a system to provide "*abundant emergency core cooling*" in event of larger breaks in the reactor coolant system. The Core Spray and Residual Heat Removal Systems at most BWRs fulfill this requirement. Because the pumps, electric motors, and piping for these systems are located within secondary containment, GDC 44 calculations and evaluations must account for heat loads from their operation during an accident. If the heat loads from core spray or RHR operation during an accident had been excluded from GDC calculations, it would be immaterial with respect to GDC 44 compliance whether the design complied with GDC 35. Both GDC must be met.

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	The underlying foundation for our petition is our contention that the heat loads from spent fuel pools and their support equipment have not been accounted for within GDC 44 calculations and evaluations. Thus, while GDC 61 is being met for the spent fuel pools inside containment, GDC 44 might not be met. Both GDC must be met. Our petition sought Demands For Information on how the applicable reactors complied with GDC 44. We assume and accept that the reactors comply with GDC 61. By whatever means are used to comply with GDC 61 (e.g., cooling by safety-related system, cooling by non-safety-related-system, non-cooled with makeup to boiling pool, etc.), heat is released into secondary containment. GDC 44 cannot be met unless that heat load, along with the heat loads from all other sources, can be removed to maintain the design basis conditions within secondary containment. The heat loads from the spent fuel pool that may be unaccounted for are not insignificant. The matter first came to my attention while working on the power uprate project for Susquehanna in the early 1990s. The GDC 44 calculation for post-accident conditions within secondary containment assumed a total heat load of around 4.5 Million British Thermal Units per hour (MBTU/hr). This calculation accounted for heat loads from HPCI, core spray, and RHR pump operation, heat losses from piping filled with warm water, and even from the heat emitted by lighting within the reactor building. But it assumed that the spent fuel pool essentially disappeared during an accident. It considered neither the latent heat from the spent fuel nor any heat released from equipment operating inside secondary containment to cool the pool's water or provide makeup to a boiling pool. The design basis heat load in each spent fuel pool at Susquehanna was 12.6 MBTU/hr while the design basis emergency heat load from a full core offload was 32.6 MBTU/hr. Thus, the GDC 44 calculation neglected heat loads 2.8 to 7.2 times greater than the total heat load considered. The results from the non-conservative GDC 44 calculation did not show abundant temperature profile margins in most reactor building areas. The Demands For Information are still needed, and still requested, to answer the vital questions of how reactors with spent fuel pools inside secondary containment comply with GDC 44.
Page 10, first full paragraph	The Proposed Director's Decision states "*The NRC staff has not identified another design basis event* [other than design bases earthquake*] that would directly result in a sustained loss of forced* [spent fuel pool] *cooling*." **UCS Comment:** The NRC staff failed to identify other design basis events that result in a sustained loss of forced spent fuel pool cooling. Among the many examples we could cite is contained within the licensee event report (LER) dated March 21, 2012, (ML12083A194) for an oil leak from an emergency diesel generator (EDG) on Browns Ferry Unit 1. The small oil leak

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	was reported because it "*did not meet the 7-day mission time of the Unit 1/2 C EDG*." Hence, the design basis for this NRC-licensed facility includes offsite power being unavailable for 7 days, a sustained period. The Residual Heat Removal (RHR) system at Browns Ferry is designed to provide forced cooling of the spent fuel pool (see letter dated December 31, 2014, ML14365A183) in fuel pool cooling assist mode. The RHR pumps can be powered from the electrical buses connected to the EDGs, and thus would be available even when offsite power was not. But not all boiling water reactors have the RHR fuel pool cooling assist mode option. And even those reactors equipped with this option may be unable to use it when necessary. The Updated Final Safety Analysis Reports (UFSARs) typically do not describe a loss of normal spent fuel pool cooling as a design basis accident or transient. Consequently, the valves that must be operated to establish the RHR fuel pool cooling assist mode are not guaranteed to be included within the periodic testing and inspection programs that provide reasonable assurance that the safety function can be performed. If RHR fuel pool cooling assist is being relied upon to mitigate design basis events, than all structures, systems, and components necessary to establish that alignment and sustain its operation must be included within appropriate testing and inspection programs. The responses to the Demands For Information sought by the petitioners would have identified what equipment was being relied upon, enabling determinations whether this equipment was adequately covered by testing and inspection programs. The NRC should issue the Demands For Information to ensure that whatever means are being used to remove heat from secondary containments are likely to perform that safety function when needed.
Page 10, first full paragraph	The Proposed Director's Decision states "*The low decay heat rate within the SFP [spent fuel pool] would require substantial time to heat the SFP coolant inventory to near saturation conditions, and restoration of the forced cooling function has a high probability of success during that time*." **UCS Comment:** What does the NRC staff mean by "*substantial time*" – a day, a week, or a month? If it involves less than 30 days, it deviates from longstanding industry and NRC practice. As stated in the licensee event report dated December 31, 2012 (ML13002A391) for Pilgrim, "*The mission time for the secondary containment system is 30 days*." As stated in the licensee event report dated September 29, 2004 (ML042810116) for Pilgrim, "*The SGTS* [standby gas treatment system*] air accumulators (accumulator bank) function* [is] *to store sufficient pneumatic energy for operation of the SGTS for the 30-day mission time*."

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	As stated in the licensee event report dated June 17, 2011 (ML11174A039) for Quad Cities Unit 1, the Residual Heat Removal Service Water System (RHRSW) has a "*mission time of 30 days*." As stated in the licensee event report dated August 15, 2007 (ML072400342) for Susquehanna, "*The design basis mission time for ESW [emergency service water] and RHRSW is defined in the FSAR as 30 days*." As stated in the licensee event report dated April 18, 2008 (ML081120106) for Hatch Unit 2, the reported problem "*could have prevented the RHRSW system from meeting its 30-day mission time*." The 30-day mission times for secondary containment, SGTS, RHRSW, and ESW all directly relate to GDC 44. In other words, one cannot comply with GDC for a "*substantial time*" that is less than 30 days. The NRC has taken enforcement action in recent years because the 30-day mission time might not have been met: By inspection report dated February 9, 2011 (ML110400431), the NRC issued a Green finding for a problem at Browns Ferry for an RHR pump motor problem that "*would have prevented the pump from performing its intended safety functions during the system's required mission time*." According to this NRC letter, "*The mission time of the 1C RHR pump to perform its intended safety functions was 30 days*." The NRC has not allowed BWR owners to only show adequate Net Positive Suction Head for emergency pumps during a "*substantial time*" of the accident: In Enclosure 1 to SECY-11-0014 dated January 31, 2011 (ML102110167), the NRC informed its Commissioners that "*The necessary time for a pump using containment accident pressure should include not only the duration of the accident when NPSH margin may be limited, but any additional time needed for operation of the pump … This additional time is usually taken as 30 days*." The NRC also applied the 30-day mission time when evaluating the potential for post-accident debris to impair emergency pump operation: NUREG/CR-7011, "Evaluation of Treatment of Effects of Debris in Coolant on ECCS and CSS Performance in Pressurized Water Reactors and Boiling Water Reactors," dated May 2010 (ML100960388) explicitly stated the NRC expectation that licensees would evaluate the postulated design basis event assuming "*All ECCE and CSS pumps are in operation for an extended period (up to the maximum mission time)…*". The NRC rejected an industry comment seeking to neglect potential damage to emergency pumps from cavitation because it would likely occur after a substantial time:

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	The nuclear industry commented on Draft Regulatory Guide DG-1234 that cavitation was a long-term degradation effect that should be excluded from post-accident assessments of emergency core cooling system (ECCS) pump performance. Its evaluation of public comments (ML111330292) stated "*The NRC staff disagrees with the comment. Cavitation over the post-LOCA mission time could affect pump performance*." Section 6.2.3, Secondary Containment Functional Design, of NUREG-0800, Standard Review Plan (ML063600406) Acceptance Criterion 1.G states "*Heat loads generated within the secondary containment (e.g., equipment heat loads) should be considered.*" There's no qualifier allowing heat loads to be ignored as long as GDC 61 is met. Hence, heat generated by the irradiated fuel in the spent fuel pools and by equipment operated in support of the spent fuel pools must be considered. Section 9.2.5, Ultimate Heat Sink, of NUREG-0800, Standard Review Plan (ML070550048) Acceptance Criterion 3.A indicates that GDC 44 is met if the design provides "*The capability to transfer heat loads from safety-related SSCs to the heat sink under both normal and accident conditions*." Secondary containment is a safety-related structure. The last paragraph on page 9.2.5-5 directs the NRC reviewers to evaluate "*the UHS design, including assumptions for heat loads*…". Compliance with GDC 44 requires a showing that heat loads within containment can be adequately removed over the entire 30-day mission time, not for a shorter period whether deemed substantial or not. As Abraham Lincoln might have observed, it is not sufficient to remove all the heat loads for some of the time, or to remove some of the heat loads all the time, but only to remove all the heat loads for the entire mission time. The Demands For Information are still needed, and requested, to indicate how licensees are complying with this regulatory requirement.
Page 10, first full paragraph	The Proposed Director's Decision states "*a sustained loss of SFP forced cooling (i.e., the heating of the SFP to the extent that the reactor building environment would be substantially changed) has not been considered among the design basis events that create a harsh environment*." **UCS Comment**: In the second full paragraph on page 9 of the Proposed Director's Decision, the NRC staff states "*the NRC staff has considered design basis high-energy line breaks, such as reactor loss-of-coolant accidents (LOCA) and steam-line break accidents, as causes of harsh environments*" for the environmental qualification requirements of 10 CFR 50.49. UCS agrees with these NRC statements – in fact, they form the basis for the request in our petition for Demands For Information on how the applicable reactors comply with 10 CFR 50.49 for design basis events.

Section	Comment
	As noted above, the mission time for many design basis accidents such as LOCA is 30-days. It is an undeniable fact that heat will be emitted from the spent fuel pool inside secondary containment throughout the mission time. Heat will also be released from equipment operating inside secondary containment to provide forced cooling of the spent fuel pool water or to provide makeup to the pool. It is UCS's contention that the calculations and evaluations that establish the temperature profiles for rooms and areas within secondary containment may not account for the heat released from the spent fuel pool and any associated equipment operation. As a result of these omissions, the environmental qualifications (EQ) of electrical equipment within secondary containment may not ensure safety functions performed by this equipment are performed throughout their entire mission times. Similar omissions of actual heat leads inside secondary containment have compromised EQ compliance in the past. For example, the licensee event report dated June 29, 1998 (Accession No. 9807070371) informed the NRC that the post-LOCA temperatures calculated in the secondary containments for Dresden Unit 2 and 3 were incorrect. The calculations had established 104°F as the post-LOCA temperature. But those calculations had assumed the post-LOCA heat loads were the same as the heat loads during normal operation. The error was in "ignoring the slow build up of temperatures in the reactor building due to the combined effect of loss of ventilation due to the post-LOCA isolation of the secondary containment and heat load generated in the reactor building due to operating equipment and lighting." When the calculations were redone to correct these omissions, the resulting temperatures in secondary containment ranged from 121°F to 152°F. Electrical equipment was replaced at Dresden to restore compliance with 10 CFR 50.49. The Demands For Information are still needed, and requested, to show how licensees comply with 10 CFR 50.49. When EQ calculations and evaluations properly include the post-accident heat loads from the spent fuel pool and any supporting equipment operation, their responses should be quite simply and straight-forward. But if the spent fuel pool and/or supporting equipment heat loads have been ignored in these calculations, the secondary containment temperatures will likely be non-conservative as they had been at Dresden.