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Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel

Comment On: NRC-2015-0106-0001
Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel; Draft NUREG for Comment

Document: NRC-2015-0106-DRAFT-0005
Comment on FR Doc # 2015-16540

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General Comment

Spent fuel dry storage systems are only as good as their weakest link. Many areas of NUREG-1927 allow for weak links. The NRC should provide aging management that is based on needs, not on what current U.S. thin canister vendors can provide. The thin canisters were not designed with aging management in mind and that is reflected in inadequacies identified in the attached document. Other dry cask designs are available that do not have the limitations of these thin canisters. This link provides information on this and related issues. Reasons to buy thick nuclear waste dry storage casks and myths about nuclear waste storage, Donna Gilmore, April 16, 2015 https://sanonofresafety.files.wordpress.com/2011/11/reasonstobuythickcasks2015-04-16.pdf

In my presentation at the November 2014 NRC Regulatory Conference on Nuclear Waste (REG CON 2014), I identified these issues and made recommendations. Apparently, those recommendations and concerns have been ignored.

Video presentation
https://youtu.be/KvAbDX0R2Eg

Slide presentation
https://sanonofresafety.files.wordpress.com/2014/10/dry-cask-storagedgilmore2014nov19.pdf

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R. ... (rdt3)

Q&A on unresolved issues
https://youtu.be/SjvJmE6ZKuM

Dr. Wolfgang Steinwartz, Executive Vice President of Siempelkamp (thick cask manufacturer), also made a presentation on his products. These casks do not have the problems of the thin canisters. Therefore, it is not necessary for the NRC to lower standards to meet U.S. aging management needs.

Dr. Steinwartz video presentation
https://youtu.be/mGJfve6ecIU

Dr. Steinwartz slide presentation
http://pbadupws.nrc.gov/docs/ML1432/ML14323A940.pdf

Specific NUREG-1927 comments are on the attached file and are also on link below.
https://sanonofresafety.files.wordpress.com/2015/08/donnagilmorecommentsnureg-1927rev1-2015-08-21.pdf

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Attachments

DonnaGilmoreCommentsNUREG-1927Rev1-2015-08-21

August 21, 2015

To: Nuclear Regulatory Commission
Carol.Gallagher@nrc.gov
Ricardo.Torres@nrc.gov

From: Donna Gilmore, SanOnofreSafety.org
dgilmore@cox.net

Subject: NUREG-1927 Rev 1 Draft, Docket ID NRC-2015-0106 Comments

Reference: NUREG-1927, Rev 1 - Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel, revised 6/29/2015
http://pbadupws.nrc.gov/docs/ML1518/ML15180A011.pdf

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Specific NUREG-1927 comments:

1. **Stress Corrosion Cracks should not be allowed without a seismic evaluation.** There should be a seismic evaluation prior to allowing any cracking in spent fuel canisters. Specifying a 75% or any other size crack for that matter, should not be allowed without a seismic analysis to determine the impact of a seismic event on a cracked canister. This could also affect transport. ASME code referenced below was **not** designed for containers filled with spent nuclear fuel. And it is not clear what justification is being used to state it is applicable for canisters filled with spent nuclear fuel.

   The NRC example on Page B-8 states: *Canisters that show evidence of localized corrosion and/or stress corrosion cracking that exceeds acceptance criteria identified in IWB-3640 [**75% crack depth of wall thickness**] are not permitted to remain in service.*

2. **Specifying crack percent when referencing ASME codes** would help with clarity, which is one of the NRC's stated goals. Here are two examples of where ASME codes have been clarified in documents.

   Example 1: IWB-3640 = 75% crack allowed
   Material Diagnostic of the Pressure Equipment in the Aspects of the New Prescriptions, Ewa Hajewska, et. al, IAE Annual Report 1999, page 32
   http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/31/056/31056998.pdf

   A fracture assessment procedure for austenitic piping according to ASME Section XI IWB-3640, Appendix C. The 1995 edition of ASME Boiler and Pressure Vessel Code has been followed together with a coming addendum where the so-called Z-factors have been revised and the restriction in maximum allowable crack depth changed from 60 to 75% of the wall thickness for welds by SMAW and SAW. Only analytical solutions according to Appendix C are included in the program.

   Example 2: NRC Safety Evaluation of a Partially Completed Weld Overlay Repair of the 02BS-F4 Weld in the Reactor Recirculation System Piping at Quad Cities Nuclear Power Station, Unit 1, January 21,2003, page 2

   http://pbadupws.nrc.gov/docs/ML0302/ML030210488.pdf

   The licensee calculated the allowable flaw depths according to (1) the limit of 75% of the pipe wall thickness, to which the ASME Section XI allows the flaw to grow

3. **Current available methods do not exist to find stress corrosion cracks or determine the depth of cracks in canisters filled with spent nuclear fuel**. This document should state this and have an aging management plan that addresses this. After attending July and August NRC meetings on stress corrosion cracking issues, it was stated the NRC is allowing the industry up to 5 years to develop technology to find and measure cracks. This document should address that and also address what the aging management plan will be if this cannot be done.

NRC 8/5/2014 stress corrosion cracking meeting summary
http://pbadupws.nrc.gov/docs/ML1425/ML14258A081.pdf

4. **Repairing canisters filled with spent nuclear fuel is currently not feasible**. First you must find the crack, in the face of millions of curies of radiation, then to repair it without introducing another corrosion factor is not feasible. One of the leading U.S. canister vendors stated this at a 2014 Southern California Edison Community meeting. Problems of attempting to repair canisters have also been referenced elsewhere. More references can be provided, if desired.

   Dr. Kris Singh, Holtec President, statements on 10/14/2014 at Edison Community Engagement Panel
   https://www.youtube.com/watch?v=euaFZt0YPi4&feature=youtu.be

   https://sanonofresafety.files.wordpress.com/2015/08/attachment-14-declaration-of-donna-gilmore.pdf

5. **No mitigation options are identified.** The mitigation statements need to be expanded to show examples of actual mitigation options. None have been provided. Maybe because no acceptable ones exist. For example, at Monticello, a canister has been sitting in a transfer cask for years because of a lack of an acceptable mitigation option.

   2014/06/04 NRC Summary of May 14, 2014 meeting with Xcel Energy Regarding Proposed Exemption Request for Dry Storage Canisters at Monticello Nuclear Generating Plant, June 4, 2014, Docket No. 72-58
   http://pbadupws.nrc.gov/docs/ML1415/ML14156A023.pdf

   Xcel Energy presented two alternatives, removing the spent fuel from the DSC or removing the outer top cover plate and reworking the welds. Both of these alternatives involve cutting the DSCs open, ***a first of a kind evolution. Xcel Energy stated that removing the inner top cover plate is a precise job and a lot of dose is expected.*** *NRC staff stated that removing the outer top cover and evaluating the welds on the inner top cover would allow Xcel Energy to gather information by inspecting the welds and would give more evidence for the exemption.* ***Xcel Energy stated that cutting the top cover plate could damage the inner top cover as well as the shell and potentially destroy the whole unit. The discussion also covered the prospects of unloading the fuel. Xcel Energy stated the latter is difficult because it is a precision job and could potentially introduce contaminates/particulates in the spent fuel pool and compromise its purity.***
   DSC 16 is in the transfer cask (TC) on the reactor building refuel floor.

6. **Appendix C: Allowing inspection of a lead canister (one canister per site or even one canister to represent multiple sites) rather than requiring inspections of all canisters results in unacceptable risks.** As the NRC stated in the August 5, 2014 stress corrosion cracking meeting previously referenced, when a crack will initiate is not predictable. We can know conditions for cracking, but not which canisters will have cracks. Two canisters loaded the same year – once could have cracks and the other may not.

   The Koeberg Plant in South Africa had a similar component (tank) that had a through wall crack in 17 years with cracks up to 0.61" deep.

Chloride-Induced Stress Corrosion Cracking Tests and Example Aging Management Program, Darrell S. Dunn, August 5, 2014
http://pbadupws.nrc.gov/docs/ML1425/ML14258A082.pdf

A Diablo Canyon canister has conditions for cracking in a 2-year old canister, even though at the 8/5/2014 meeting the NRC thought cracking wouldn't initiate for 30 years, because they assumed canisters would not be cool enough for salts to deliquesce (dissolve) on the canister. At this meeting, the NRC staff said after initiation, a crack could go through the wall of the canister in 16 years.

Diablo Canyon: conditions for stress corrosion cracking in 2 years, D. Gilmore, October 23, 2014
https://sanonofresafety.files.wordpress.com/2011/11/diablocanyonscc-2014-10-23.pdf

Canisters have been loaded at San Onofre since 2003. It is located in a similar environment to Koeberg – onshore winds, surf, and frequent fog. I do not see a solution in NUREG-1927 that will adequately address this real-life example.

7. **Concrete inspection and repair guidelines assume both of these can be done to an acceptable degree, but there is lack of data to support this.** A recent two-day NRC concrete workshop with numerous concrete experts identified numerous potential structural degradations and the lack of inspection capabilities. Unless these tools can be identified, establishing adequate aging management for inspection and repair appears more wishful thinking than reality.

   NRC's Expert Panel Workshop on Degradation of Concrete in Spent Nuclear Fuel Dry Cask Storage Systems, February 24-25, 2015, identified numerous concrete aging management problems, particularly with below ground systems (such as the Holtec UMAX dry storage system) due to limited inspection capability, ground moisture and chemical reactions with concrete. The NRC's solution is to lower standards and require less frequent inspections, as stated on Page B-14:

   > ***For visual inspections, the frequency of inspection is defined as:***
   > • *For above-grade (accessible and inaccessible) areas: ≤ 5 years*
   > • *For below-grade (underground) areas: ≤ 10 years, and when excavated for any reason…*
   >
   > And Page B-16 Concrete
   > *For the groundwater chemistry program, the acceptance criteria are commensurate with ASME Code Section XI, Subsection IWL, which states that an aggressive below-grade environment is defined as:*
   > • *pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm.*

In the Concrete Workshop, a concerned was raised that removing the soil on the sides of the underground system could result in instability. I realize this was just the start of exploring concrete aging issues. However, this NUREG assumes solutions exist.

Concrete is not an issue in thick ductile cast iron casks, since they don't use concrete for gamma and neutron shielding.

NRC Concrete Expert Panel Workshop, February 24-25, 2015
Agenda (ML15036A603)
http://pbadupws.nrc.gov/docs/ML1503/ML15036A603.pdf

Slide presentation, February 24-25, 2015 (ML15051A369)
http://pbadupws.nrc.gov/docs/ML1505/ML15051A369.pdf

Transcript February 24, 2015 (ML15093A003)
http://pbadupws.nrc.gov/docs/ML1509/ML15093A003.pdf

Transcript February 25, 2015 (ML15093A004)
http://pbadupws.nrc.gov/docs/ML1509/ML15093A004.pdf

8. **Implementation of AMP(s) should be prior to CoC renewal**. Given the many unknowns regarding inspection, repair and mitigation solutions, this statement on Page 39 allowing CoC renewal before implementation of AMP(s) should be deleted. *"Generally, development of the infrastructure for AMP implementation should be no later than one year from the date the NRC issues a renewed specific license or CoC. However, in some situations, shorter or longer AMP implementation periods may be appropriately justified."*

9. **Appendix B Examples of Aging Management Program identify inspection options, but do not mention that most of these are not currently available for canisters filled with spent nuclear fuel.** These should be clarified to indicate which could actually be used today and when these might be available, and what the alternative will be if they are not available. In addition, a comment on Page B-3 and other pages allows exclusion for non-accessible areas. Considering a large part of the thin canisters are currently not accessible, this makes for a large and unacceptable weak link in aging management: *"For accessible areas where adequate cleaning can be performed, remote visual inspection meeting the requirements for VT-1"*

10. **Statements about incidence of CISCC are misleading at best and should be clarified.**

    a. The statement on page B-9 "*No cases of CISCC for stainless steel dry storage canisters have been reported*" should have the qualifier, "since these canisters have not been installed long enough for most of them to experience CISCC" and none of them have been inspected due to limited technology.

    b. The page B-9 statement "*Inspections of dry storage canisters after 20 years in service have been conducted at a few independent spent fuel installation (ISFSI) sites*", should be clarified to include the limitations of that "inspection". It was not an inspection of stress corrosion cracks and in no way indicates whether there are cracks on those canisters. It was a very limited check of temperature on parts of a few canisters and a limited check for surface contaminants and a limited visual check for other corrosion.

c. The page B-9 statement should be expanded to include the aforementioned EPRI Diablo Inspection results and the limitation of the inspections. Instead it only states: "*Details of the inspection conducted at the Calvert Cliffs nuclear power plant ISFSI are documented in a recent EPRI report (Waldrop et al., 2014). No evidence of localized corrosion was identified but some amount of chloride-containing salts were determined to be present and corrosion products believed to be related to iron contamination were identified.*"

11. The following corrective actions (Page B-7 and B-8) cannot be implemented until the previously discussed items are addressed.

Corrective Actions
Disposition of Canisters with Aging Effects.

For austenitic stainless steel canisters covered by an AMP that utilizes the inspection and acceptance criteria in ASME B&PV code Section XI for Class 1 piping system, the disposition of canisters should be commensurate with in-service inspection results:

- *Canisters with no evidence of corrosion are permitted to remain in service and will continue to be inspected at 5-year intervals.*
- *Canisters with rust deposits that are determined to be a result of iron contamination but do not have evidence of localized corrosion or stress corrosion cracking are permitted to remain in service and will continue to be inspected at 5-year intervals.*
- *Canisters that show evidence of localized corrosion and/or stress corrosion cracking that does not exceed the acceptance standards in IWB-3514.1 are permitted to remain in service and will be inspected at 5-year intervals. Sample size will be increased to assess 25 percent of canisters with similar time in service (± 5 years) or a minimum of one additional canister with a time in service closest to the original sample within one year of the completed in-service inspection date. Results of the initial inspection and the schedule for additional inspections will be reported to the NRC. In addition, the results for the additional in-service inspections will be reported to the NRC upon completion.*
- *Canisters that show evidence of localized corrosion and/or stress corrosion cracking that exceeds the acceptance standards in IWB-3514.1 but meet the acceptance criteria identified in IWB-3640 are permitted to remain in service and will be inspected at 3-year intervals. Sample size will be increased to assess 50 percent of canisters with similar time in service (± 5 years) or a minimum of one additional canister with a time in service closest to the original sample within one year of the completed in-service inspection date. Results of the initial inspection and the schedule for additional inspections will be reported to the NRC. In addition, the results for the additional in-service inspections will be reported to the NRC upon completion.*
- *Canisters that show evidence of localized corrosion and/or stress corrosion cracking that exceeds acceptance criteria identified in IWB-3640 are not permitted to remain*

in service. Upon identification, the in-service inspection sample size will be increased to assess 100 percent of canisters with similar time in service (± 5 years) or a minimum of one additional canister with a time in service closest to the original sample within one year of the completed in-service inspection date. Results of the initial inspection, the schedule for mitigation either by repair or replacement and the schedule for additional inspections will be reported to the NRC. In addition, the results for the additional in-service inspections will be reported to the NRC upon completion.

12. The section on High Burnup Fuel (Page B-19) is slanted by only mentioning studies that support the hopeful conclusion that the fuel will not degrade after storage. It should be more balanced and address studies that show the opposite and provide aging management for that situation.

    These Billone papers show newer Zirconium alloy claddings (Zirlo and M5) degrade faster with high burnup fuel than earlier claddings, such as Zircaloy-4.

    Ductile-to-Brittle Transition Temperatures for High-Burnup PWR Cladding Alloys, Mike Billone and Yung Liu Argonne National Laboratory U.S. NWTRB Winter Meeting November 20, 2013, DOE Slide Presentation
    http://www.nwtrb.gov/meetings/2013/nov/billone.pdf

    Embrittlement and DBTT of High-Burnup PWR Fuel Cladding Alloys, FCRD-UFD-2013-000401, Billone, et.al, September 30, 2013
    https://sanonofresafety.files.wordpress.com/2014/02/billone2013-09-30embrittlementdbtthighbrnup-pwrfuelclad-alloys.pdf

    Ductile-to-Brittle Transition Temperature for High-Burnup Zircaloy-4 and ZIRLO™ Cladding Alloys Exposed to Simulated Drying-Storage Conditions M.C. Billone, T.A. Burtseva, and Y. Yan Argonne National Laboratory September 28, 2012.
    http://pbadupws.nrc.gov/docs/ML1218/ML12181A238.pdf

    "...the trend of the data generated in the current work clearly indicates that failure criteria for high-burnup cladding need to include the embrittling effects of radial-hydrides for drying-storage conditions that are likely to result in significant radial-hydride precipitation...A strong correlation was found between the extent of radial hydride formation across the cladding wall and the extent of wall cracking during RCT [ring-compression test] loading."