CHAIRMAN Resource

From: Donna Gilmore [redacted]

Sent: Wednesday, March 13, 2019 11:25 PM

To: RulemakingComments Resource; Allen, William; Lohr , Edward

Cc: Mike Levin; Hutt, Heather (Harris); Jeff McDonald; Teri Sforza; Joseph Street; Toni Iseman; Decommission; CHAIRMAN Resource; CMRBARAN Resource; CMRCaputo Resource; CMRBurns Resource; Ace Hoffman; Len Hering; Gregory Jaczko; Tom English; Subrata Chakraborty; Bart Ziegler; Layton, Michael

Subject: [External_Sender] NUHOMS canister system CoC 1029, Amendment 4 Docket ID NRC-2018-0265 Areva TN Americas Donna Gilmore comments

Ms. Gilmore,

Thank you for comments on the "List of Approved Spent Fuel Storage Casks: TN Americas LLC Standardized Advanced NUHOMS® System, Certificate of Compliance No. 1029, Amendment No. 4," rulemaking received by the NRC on March 13, 2019 (NRC ADAMS Accession Number ML19073A105). Although the public comment period for the rulemaking closed on January 28, 2019, the NRC staff reviewed your comments. Based on the review of the comments, the NRC staff determined that they were not substantive and would not have changed the NRC's decision on issuing the rule. Specific responses to the comment are provided in the following paragraphs.

1. **License Amendment 4 will eliminate the requirement to report peak radiation levels from the outlet air vents of the Areva TN Americas NUHOMS canisters at San Onofre and elsewhere. Through-wall cracks in canisters will have highest radiation readings from the outlet air vents`. What is the purpose of not reporting peak levels from the outlet air vents other than to hide high radiation levels?**

   I have requested the NRC provide radiation levels of the outlet air vents at San Onofre. The NRC has repeatedly refused to share the outlet air vent radiation levels of the NUHOMS canisters that are up to 15 years old. Why? Are they already leaking? What are you hiding?

   Response: The history of the dose rate measurements for the CoC 72-1029 and the Amendments to the CoC are provided in the subsequent paragraphs. The CoC was originally approved on February 2, 2003. Amendments to the CoC include Amendments 1 (approved May 31, 2005), 3 (approved February 20, 2015) and 4 (approved February 5, 2019). The CoC holder withdrew the application for Amendment 2.

The only changes to CoC 1029 Technical Specification Section 5.2.4 from the original CoC (ML030100468) have been the addition of required dose rate limits for DSCs and transfer casks as new systems were added to the CoC.

The Technical Specifications for CoC 1029 (ML030100468) and Amendment 1 of the CoC 1029 (ML051520131) did not include dose rate measurements requirements for the exterior of the Advanced Horizontal Storage Module (AHSM) loaded with either a 24PT1 or a 24PT4 dry shielded canister (DSC).

Amendment 3 of the CoC 1029 added the Advanced Horizontal Storage Module for High Burnup Fuel and High Seismic (AHSM-HS) and the 32PTH2 DSC. Additional technical specifications were included with the addition of the 32PTH2 DSC and the AHSM-HS. These additional technical specifications included the requirement for dose rate measurements at the AHSM-HS inlet bird screen and at the outside surface of the AHSM-HS door. The dose rate measurement requirements and dose rate limits are provided in CoC 1029 Amendment 3 Technical Specifications section 5.4.2 (ML15054A513).

Amendment 4 of CoC 1029 added the requirement to measure dose rates for the Advanced Horizontal Storage Module loaded with either a 24PT1 or a 24PT4 DSC. Additional clarifications were included on the locations of the dose rate measurements but the dose rate limit and the number of measurements required for the AHSM-HS with a 32PTH2 DSC were not changed. The dose rate limits for the AHSM with either a 24PTH1 or a 24PTH4 DSC and the dose rate limits for the AHSM-HS with a 32PTH2 DSC are provided in CoC 1029 Amendment 4 Technical Specifications section 5.4.2 (ML19036A558).

Any general licensee using the CoC 1029 system must comply with the CoC Technical Specification Section 5.2.3 Radiological Environmental Monitoring Program, which requires (a) the implementation of a program to ensure that the annual dose equivalent to an individual located outside the ISFSI controlled area does not exceed the annual dose limits specified in 10 CFR 72.104(a); and (b) operation of the ISFSI will not create any radioactive materials or result in any credible liquid or gaseous effluent release.

Also, a general license using the CoC 1029 system must comply with the CoC Technical Specification Section 5.2.4 Radiation Protection Program and establish administrative controls to limit personnel exposure to As Low As Reasonably Achievable (ALARA) levels in accordance with 10 CFR Part 20 and Part 72. Pursuant to 10 CFR 72.212, the general licensee must perform an analysis to confirm that the limits of 10 CFR Part 20 and 10 CFR 72.104 will be satisfied under the actual site conditions and configurations considering the planned number of DSCs to be used and the planned fuel loading conditions. In addition, the general licensee must have a monitoring program to ensure the annual dose equivalent to any real individual located outside the ISFSI controlled area does not exceed regulatory limits. The Radiological Environmental Monitoring Program is included in CoC Technical Specification Section 5.2.3.

2. **The NUHOMS canisters at Calvert Cliffs in Maryland are up to 27 years old. They already have approval to hide peak radiation levels from the outlet air vents. Are those already leaking? What are the radiation levels from the outlet air vents of those canisters?**

   Response: As a specifically licensed ISFSI under 10 CFR Part 72, the Calvert Cliffs ISFSI must comply with the regulatory requirements for dose rate limits to any individual located on or beyond the nearest boundary of the controlled area specified in 10 CFR Part 72.104 and 72.106. The Calvert Cliffs Nuclear Power Plant (CCNPP) has thermoluminescent detectors (TLDs) positioned on the ISFSI perimeter fencing that are read semiannually to provide a record of radiation exposure at the ISFSI perimeter. In addition, CCNPP has an environmental monitoring program that includes dosimeters, air samplers, and vegetation and soil samples at the ISFSI site.

3. **The NRC on August 5, 2014 stated once cracks start in canisters they can grow through the wall in 16 years. At the time they said it would be 30 years before canisters would be cool enough for moist salt air to start the corrosion cracking process. However, with the EPRI investigation of a 2-year old Diablo Canyon canister that already had a low enough canister temperature and corrosive salt particles, you know your 30 year estimate is wrong. Why do you continue to ignore this issue?**

   Response: The NRC staff has not ignored this issue. Specific answers to these and similar questions have been provided multiple times including published responses to public comments on NUREG-1927 Revision 1 (ML16125A534) and ISG-2 Revision 2 (ML16117A082). Specific responses are provided below.

   August 5, 2014 NRC Meeting Summary

   https://pbadupws.nrc.gov/docs/ML1425/ML14258A081.pdf

   The August 5, 2014 NRC/NEI chloride inducted stress corrosion cracking (CISCC) Regulatory Issue Resolution Protocol (RIRP) meeting did not include an assessment of potential canister crack growth rates. The meeting was held to discuss the progress towards understanding CISCC and developing approaches to predict susceptibility and growth rates.

   The NRC staff comment was based on calculated CISCC growth rates using the Arrhenius equation (i.e., crack growth rate was function of temperature). The NRC staff had not paired the Arrhenius equation that included parameters for CISCC as a function of temperature with environmental parameters that were required to model crack growth rates. At the time, the data available to develop a temperature dependent crack growth rate equation was limited and the uncertainty of the available data and operational experience of CISCC events were not quantified. Nevertheless, the NRC staff needed to assess the minimum inspection interval for aging management program (AMP) inspections for upcoming 10 CFR Part 72 renewals. To address the uncertainty in the available data, the NRC used a conservative assumptions that included a temperature of

50°C (122°F) and the assumption that the conditions for CISCC to occur would be continuously present. In reality, these conditions occur infrequently and are typically limited to a few hours per day over a duration of a few days. These conservative assumptions resulted in a calculated crack growth rate slightly greater than 0.8 mm/year (0.03 in/year) which would penetrate 13 mm (0.5 in) stainless steel section in a period of 16 years.

During the next CISCC RIRP meeting on April 21, 2015, EPRI (NRC ADAMS package ML15142A805) presented the results of their work published in, "Flaw Growth and Flaw Tolerance Assessment for Dry Cask Storage Canisters," EPRI-3002002785, Palo Alto, CA: 2014. The NRC independently developed an CISCC model which also used data from the National Oceanic and Atmospheric Administration (NOAA) coupled to the temperature dependent crack growth rate calculation that the NRC presented at the August 5, 2014 CISCC RIRP meeting. Although the EPRI and the NRC temperature dependent crack growth rate calculations were based on different sets of data, the results of the CISCC assessments were similar. In a hot and humid environment such as Florida, the conditions for CISCC are more prevalent compared to the milder California coastal environment. Using the developed CISCC model, through wall CISCC in Florida was determined to be possible in approximately 26-30 years whereas in California, through wall CISCC was calculated to take approximately 80 years.

A plot of the crack growth rate as a function of temperature and the sources of data (Kosaki 2008; Hayashibara et al. 2008) for the Arrhenius equation parameters is shown on slide 4 of the NRC presentation included in ML15146A115. These are the same Arrhenius equation parameters used in the August 5, 2014 CISCC RIRP meeting. As stated in slide 11, the model still included multiple conservative assumptions which likely over predict CISCC growth rates resulting in shorter calculated times for through wall cracking.

Diablo Canyon: conditions for stress corrosion cracking in 2 years

Conditions necessary for CISCC are not present on the canisters stored at the Diablo Canyon ISFSI. The NRC staff are aware of a publicized assertion regarding the Diablo Canyon ISFSI made by a member of the public October 23, 2014, "Diablo Canyon: conditions for stress corrosion cracking in 2 years." **https://sanonofresafety.files.wordpress.com/2011/11/diablocanyonscc-2014-10-23.pdf**

This assertion assumes unrealistic humidity values, inaccurate information regarding both the composition of the dust deposits on the Diablo Canyon canisters and conditions for deliquescence resulting in an improbable assessment of crack initiation and crack growth rate.

The staff considers the assumed absolute humidity values of 40-45 g/m^3 are unrealistic based on the range of observed atmospheric weather conditions. An absolute humidity of 40 g/m^3 would result in a dew point of ~37° (98 °F) which is greater than any recorded dew point measured in the U.S. (such conditions would be immediately life threating to

the general population). Ambient absolute humidity values above 30 g/m^3 are rare and are typically limited to a few hours per day over a duration of a few days.

CISCC at 85°C (185°F) by the deliquescence of $MgCl_2$ (conservatively using a relative humidity of 18% which was the lowest measured relative humidity for efflorescence of sea salt at 80°C (176°F) reported by He et al., 2014) would only be possible in an ambient environment with an absolute humidity of 63 g/m^3 corresponding to a dew point of 46°C (115°F). In order for such conditions to continuously occur, the ambient low temperature during the diurnal cycle must continuously be at or above 46°C (115°F). Such conditions have never been observed in an ambient environment.

The assessment of the conditions on the Diablo Canyon and Hope Creek canisters and the analyses of samples obtained from the surfaces of canisters at these ISFSI are described in SANDIA REPORT SAND2014-16383, "Analysis of Dust Samples Collected from Spent Nuclear Fuel Interim Storage Containers at Hope Creek, Delaware, and Diablo Canyon, California," Albuquerque, New Mexico: Sandia National Laboratories, July 2014 (Bryan and Enos, 2014). As described in the analytical results reported by Bryan and Enos (2014), the deposited material on the Diablo Canyon canisters is sodium chloride (NaCl) with some magnesium sulfate ($MgSO_4$). No magnesium chloride ($MgCl_2$) was found on the Diablo Canyon canisters. The conclusion section of Bryan and Enos (2014) report explains the limitations for the sample collection and analyses.

Bryan, C.R. and D.G. Enos. SAND2014-16383, "Analysis of Dust Samples Collected From Spent Nuclear Fuel Interim Storage Containers at Hope Creek, Delaware, and Diablo Canyon, California." Albuquerque, New Mexico: Sandia National Laboratories, July 2014. https://prod-ng.sandia.gov/techlib-noauth/access-control.cgi/2014/1416383.pdf.

He, X., et al., "Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts," NUREG/CR-7170, U.S. Nuclear Regulatory Commission, February 2014, ADAMS Accession No. ML14051A417.

4. **Amendment 4 will eliminate the requirement for a spent fuel pool once pools are empty. Why isn't there a requirement to have a hot cell on-site or a pool on-site in case of need to replace canisters? To assume there will not be through-wall cracks or other failures of the canisters or a need to repackage fuel for transport is not justified. As you know, either a pool or hot cell is required for these conditions.**

Response: CoC 72-1029 Amendment 4 does not eliminate the requirement for a spent fuel pool. CoC 72-1029 Amendment 4 includes revised procedures that address the eventual decommissioning of the spent fuel pool. CoC 1029 UFSAR Revision 9 Section 11.2.5.4 Corrective Actions states:

The recovery operations listed in this section assume the cask drop occurs during initial transfer and loading of the DSC into the AHSM, when the spent fuel pool is still operational and available. If a drop of the transfer cask with a loaded DSC occurs during

transfer to a transportation cask and an inspection determines that the DSC is damaged and a spent fuel pool is not available onsite, the DSC shall be placed into a safe condition. If required, the DSC could be transported offsite to a site licensed for either dry or wet unloading of the DSC.

This UFSAR revision is applicable to the 24PT1, 24PT4 and the 32PTH2 DSCs. Enclosure 3 to E-53756 Replacement Pages for ANUH-01.0150, Standardized Advanced NUHOMS® UFSAR, Revision 9 (Public Version) (ML19073A200) is publicly available in ADAMS.

NRC licensees are required to maintain systems to comply with NRC regulations, system technical specifications and NRC license conditions. If safety issues are identified with a spent fuel storage system, the licensee must pursue corrective actions to ensure that the spent fuel is safely stored. These actions would not necessarily involve replacement of major dry storage system components (e.g., canister or cask) or repackaging the spent fuel in a new system. Corrective actions would more likely include further assessment and inspection, in-place repairs to components similar to those that have actually been used on components in commercial nuclear power reactors such as the application of remote repair welding techniques or creating a secondary confinement boundary for the spent fuel if needed (e.g., nesting a canister within another container).

The NRC does not prescribe how licensees would take corrective action with respect to specific spent fuel canister designs. The NRC evaluates whether the corrective action taken is effective and sufficient to maintain the intended functions of the important-to-safety structures, systems, and components, and remain compliant with the requirements in 10 CFR Part 72. Proposed repair methods require demonstration and compliance with an NRC-approved quality assurance program.

In the event that a storage canister must be unloaded, procedures for removing fuel from welded stainless steel canisters are included in operational procedures of licensed designs. These procedures are included in the dry storage system Safety Analysis Report. These procedures have been reviewed and approved by the NRC. Per the regulatory requirements in 10CFR 72.236(h), spent fuel storage systems must be compatible with wet or dry spent fuel loading and unloading facilities. Performing such an activity should not be undertaken unless there is a specific safety need, based on indications that the canister is not performing adequately and only after evaluating other measures to remedy the circumstance with the canister along with the potential risks such activities, including opening the canister, could present.

5. The Nuclear Waste Policy Act requires the ability to retrieve fuel assemblies for the purpose of transporting the fuel to another site by the DOE. Why are you eliminating the only method to comply with this law?

Response: The design, construction and operation of the 72-1029 system is compliant with the NWPA and the NRC regulatory requirements with respect to retrievability in 10 CFR Part 72.236(m).

The DSCs included in the 1029 CoC, as well as most of the welded stainless steel canisters used in dry storage systems, are designed to be transportable inside a specially designed transportation overpack. This allows the fuel to be stored and transported without the need for additional fuel handling. The welded stainless steel canisters are leak tested prior to being put into service. This assures that the inert helium environment will be maintain inside the canister. The inert environment prevents degradation of the stored spent fuel and eliminates the need to inspect the fuel or the interior of the canister. However, if there is a safety need to open a welded canister, there is a procedure in the Safety Analysis Report which has been reviewed and approved by the NRC.

The Nuclear Waste Policy Act (NWPA) does not require the ability to retrieve fuel assemblies for the purpose of transporting the fuel to another site by the DOE. The NWPA addresses transportation of spent nuclear fuel in Title I—Disposal and Storage of High-Level Radioactive Waste, Spent Nuclear Fuel, and Low-Level Radioactive Waste, Subtitle H-Transportation. NWPA Title I, Subtitle H, Section 180(a) states that no spent nuclear fuel or high-level radioactive waste may be transported by or for the Secretary (DOE) under Subtitle A (Repositories for Disposal of High-Level Radioactive Waste and Spent Nuclear Fuel) or under Subtitle C (Monitored Retrievable Storage) except in packages that have been certified for such purpose by the Commission (NRC). The NRC's transportation regulations are in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." The NRC has certified systems for the transportation of spent nuclear fuel that use loading and unloading of individual fuel assemblies as well as systems for transportable storage canisters that do not require handling of individual fuel assemblies.

The regulatory requirements for retrievability are in 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste." Retrievability is specifically mentioned in 10 CFR 72.122(l), which states that "storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related greater than class C waste for further processing or disposal." 10 CFR 72.236(m) states that certificate of compliance (CoC) holders should design for retrievability; "[t]o the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy."

Interim Staff Guidance (ISG) – 2 Revision 2 (ML16117A080) provides guidance to the staff for determining whether an application submitted under 10 CFR Part 72, sufficiently

demonstrates that the system is designed to allow ready retrieval of spent fuel. ISG-2, Revision 2 defines ready retrieval as "the ability to safely remove the spent fuel from storage for further processing or disposal." In order to demonstrate the ability for ready retrieval, a licensee should demonstrate it has the ability to perform any of the three options below. These options may be utilized individually or in any combination or sequence, as appropriate.

A. remove individual or canned spent fuel assemblies from wet or dry storage,
B. remove a canister loaded with spent fuel assemblies from a storage cask/overpack,
C. remove a cask loaded with spent fuel assemblies from the storage location.

Public comments on ISG-2 Revision 2 and the NRC's responses to the comments received are publicly available in ML16117A082.

6. **These Sierra Club comments to the NRC busted the NRC false assumptions that nothing can go wrong once fuel is in dry storage. Why are you ignoring them? Sierra Club comments to NRC proposed rule for regulatory improvements for decommissioning power reactors, Docket NRC-2015-0070, March 2016 http://www.nrc.gov/docs/ML1608/ML16082A004.pdf**

Response: Many of the subjects included in Susan Corbett's letter regarding Docket ID NRC-2015-0070 Advanced Notice of Proposed Rulemaking (ANPR): Regulatory Improvements for Decommissioning Power Reactors Comments have previously been addressed by the NRC as responses to public comments including (1) inspection of dry storage systems; (2) the potential for aging effects including chloride induced stress corrosion cracking (CISCC); (3) security and protection against terrorism; (4) potential pyrophoricity of spent fuel storage systems contents such as zirconium hydrides; (5) risks of dry storage of spent nuclear fuel. Many of these issues were addressed in the published responses to public comments on NUREG-1927 Revision 1 (ML16125A534) and ISG-2 Revision 2 (ML16117A082). The NRC staff have not ignored these comments. See the link: https://www.regulations.gov/docket?D=NRC-2015-0070

The comments from Ms. Corbett were on the advance notice of proposed rulemaking and, while the staff did not prepare an explicit response to those comments, there is a general discussion of comment themes and NRC staff response in the final regulatory basis document (ML17215A010). The current NRC staff position on various topics in the proposed rule package which is before the Commission (ML18012A019).

The staff considered all comments received and used them to develop the proposed rule package but will not be providing detailed responses. The staff will consider comments on the proposed rule (when it is published in the Federal Register), and the staff will provide responses to significant public comments in the final rule package.

7. **Amendment 4 eliminates the requirement of daily visual inspections of air vents. Why are you only requiring temperature monitoring? Previously, you required both. Why did you previously require both, but now you don't? Why are you reducing safety?**

   Response: Temperature monitoring is an acceptable method to demonstrate the storage system vents are free from blockage allowing decay heat removal. Use of a temperature monitoring program instead of a visual inspection program does not result in a reduction of safety.

   CoC Technical Specification Section 5.2.5 addresses the AHSM/AHSM-HS Thermal Monitoring Program. Amendment 4 allows the use of temperature monitoring programs for the AHSMs loaded with the 24PT1 DSC.

   In the original CoC Technical Specification (ML030100468) visual inspection of the AHSM air vents was required. In Amendment 1 of the CoC 1029, the CoC Technical Specification (ML051520131) was revised to include requirements for the AHSM air vents for the 24PT4 DSC. The air vent monitoring program credited the temperature measurements obtained in the thermal monitoring program for the AHSMs loaded with the 24PT4 DSCs. Similarly, the 32PTH2 DSCs and the AHSM-HS were added to the CoC in Amendment 3. The CoC Technical Specification (ML15054A513) also credited the temperature measurements obtained in the thermal monitoring program for the AHSM-HSs loaded with the 32PTH2 DSCs. Amendment 4 of the CoC 1029, the CoC Technical Specifications revised the requirements for the AHSMs loaded with 24PT1 DSCs to credit the temperature measurements obtained in the thermal monitoring program. As stated in the Technical Specifications, visual inspection of all AHSMs will be used if the temperature measurements from the thermal monitoring program are not available.

8. **This approval allows increased temperatures. Why are you allowing increased temperatures? What was the reason you required lower temperatures previously? Why are you reducing safety? At the NRC RIC Conference on March 12, 2019, during the Q&A for NRC Chairman Svinicki, it was mentioned that one of the improvements NRC employees would like to see is for their management to have the courage to do the right thing. Please heed your employees' requests for our safety and yours by denying license Amendment 4 and amending any others you have approved that lowered these safety standards.**

   Response: The allowable temperatures of important to safety structures, systems and components were not increased in Amendment 4 of the 1029 CoC. Allowable temperatures of the important to safety structures systems and components including the 24PT1, 24PT4 and the 32PTH2 DSCs as well as the AHSM and AHSM-HS are provided in publicly available versions of the UFSARs. The NRC ADAMS Accession numbers for the 72-1029 FSARs are listed below.

   CoC 72-1029 FSAR: ML031040379, ML050410252, ML031040379 and ML031040312

CoC 72-1029 Amd 1 FSAR: ML17167A234 and ML17167A241

CoC 72-1029 Amd 3 FSAR: ML16228A017

CoC 72-1029 Amd 4 FSAR: ML19073A204

9. **Also, please see petition to recall the Holtec canister system at San Onofre. The system is a lemon and must be replaced. Holtec's lack of a precision downloading system unavoidably causes the walls of all canisters to be damaged the entire length of the canister wall as they are downloaded into the storage holes. Michael Layton confirmed the NRC approved this system and told me he didn't know whether the NRC knew that the Holtec system was not a precision downloading system. All the above-ground Holtec systems may likely have a similar problem, based on my research of the HI-STORM 100 FSAR. The NRC needs to reevaluate their policy of giving exemptions to destroy spent fuel pools. Rejecting Amendment 4 would be a good start. https://www.change.org/p/san-onofre-nuclear-waste-recall-defective-storage-system**

   Response: The petition has been reviewed.

From: [redacted]
To:
Cc:
Subject: Reply to email questions
Date: Monday, July 29, 2019 1:24:00 PM

Ms. Gilmore,

Thank you for your emails to the NRC Chairman and Commissioners regarding the NRC Webinar on March 25, 2019 and the use of the Holtec UMAX system at the SONGS Independent Spent Fuel Storage Installation (ISFSI). The NRC strives to establish and maintain openness in transparent and forthright communications with the public and all stakeholders. This email serves as a consolidated response to your emails dated March 25, April 9, and June 4, 2019.

Specific information regarding the Holtec HI-STORM UMAX system and the August 3, 2018 incident at the SONGS ISFSI (issues raised in several of the emails) is provided in detail below, including a summary of recent developments and available information to date, with the NRC's Agencywide Documents Access and Management System (ADAMS) accession numbers provided for information that is publicly available. Additionally, specific information regarding the more general concerns raised regarding spent fuel storage is also provided below. This information is organized to respond to each of the concerns raised in your April 9, 2019 email, which appears to provide the most comprehensive compilation of concerns.

Information regarding Holtec and incident at SONGS ISFSI:

Southern California Edison (SCE) investigated the August 3, 2018 incident that involved the lowering of the Holtec HI-STORM UMAX MPC at the SONGS ISFSI. A summary of the SCE investigation including a description of the event, a safety analysis of a potential MPC drop, the causal analysis of the event, and development and implementation of corrective actions to prevent recurrence was presented at a Pre-Decisional Enforcement Conference on January 24, 2019 (ADAMS Accession No. ML19023A033). Additional information on the incident and the corrective actions taken by SCE are described in SCE's Reply to Notice of Violation EA-18-155 (ADAMS Accession No. ML18362A148).

In addition to the analyses and the corrective actions, SCE also conducted an inspection of MPCs after downloading into the UMAX system. SCE's conclusion regarding the inspection of the canisters is included in their letter to the NRC dated May 19, 2019, regarding the resumption of fuel transfer operations at SONGS (ADAMS Accession No. ML19141A049). A description of the system used to inspect the MPCs and the results of the inspections are described in the SCE Canister Downloading Update presentation available on the SONGS community engagement panel website: https://www.songscommunity.com/community-engagement/meetings/community-engagement-panel-meeting-20190111

The NRC staff have been informed of the results of the inspection of the MPCs. The NRC staff have previously visited the Holtec fabrication facility where the MPC laser peening operation is conducted. The NRC staff reviewed the laser peening characterization and testing information, laser peening procedures, and witnessed actual laser peening of an MPC. Based on the evaluation of the laser peening process, the depth of compressive residual stresses in the welded regions of the MPC extend well beyond the depth of any scratches in the MPC that may occur during MPC downloading operation in the Holtec UMAX system. Consequently, any scratches in welded regions of the MPC would not become initiation sites for cracking, including chloride induced stress corrosion cracking (CISCC). , Scratches on portions of the MPC that do not include welds would also not result in cracking because there are no cyclic,

thermal or residual stresses present to initiate fatigue or CISCC.

Information regarding concerns presented in April 9, 2019 email

1. *Please provide the specific ASME nuclear pressure vessel codes or other ASME codes you are referring to in your video below regarding the conditions for allowing scratches or abrasions in these Holtec thin-wall nuclear pressure vessel canisters. The NRC grants exemptions to many ASME codes. In fact, these Holtec nuclear pressure vessels do not have ASME N3 stamps, as you inferred was probable, in the last San Onofre Community Engagement Panel meeting. They don't even have pressure monitoring or pressure relief valves which, as you mentioned at the CEP meeting, even our hot water heaters have ASME pressure vessel stamps. Without those features there is no ability to monitor for hydrogen gas buildup or to release the gas in order to prevent the canisters from exploding.*

   Response: The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV) code NB-3213.11 Peak Stress is located in ASME B&PV Code Section III, Subsection NB, Article 3000. ASME B&PV Code NB-3213.11, which addresses peak stress, identifies local discontinuities that may be a source of fatigue cracking or brittle fracture. In this context, the scratches on the UMAX canisters are considered local discontinuities. Fatigue is not a concern due to the absence of meaningful cyclic stresses for the spent fuel storage canister. Brittle fracture is not a concern because of the toughness of austenitic stainless steel.

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   A list of ASME Code alternatives are included in the Certificate of Compliance Appendix B Table 3-1 which is publicly available in ADAMS (ADAMS Accession No. ML16341B107).

2. *Hydrogen gas buildup is caused from the residual water remaining in these canisters after drying. No one knows how much water or moisture is in these canisters, since they are welded shut. The Nuclear Waste Technical Review Board is concerned about this issue for both defense and commercial spent nuclear fuel waste containers. See their December 2017 report to Congress and the DOE Secretary regarding Spent Nuclear Fuel Storage and Transport.*

   The NWTRB recommends all spent nuclear fuel and its containment must be retrievable, maintained and monitored in a manner to prevent radioactive releases and hydrogen gas explosions. The NRC continues to approve thin-wall welded canisters that do not meet these basic and critical safety requirements.

   Response: Dry storage for spent nuclear fuel from commercial power reactors includes procedures for removing water and drying the contents of the canister prior to backfilling the canister with helium (an inert gas) and performing a helium leak test. The Holtec canisters used in the multipurpose canister (MPC) 72-1040 system, including the canisters at the SONGS ISFSI, are individually tested to “leak tight” criteria per American National Standards Institute (ANSI) Standard N14.5, “American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment.” These procedures assure that the free water is removed, and that the contents are sufficiently dry and enclosed in an inert environment which prevents degradation to the stored contents.

The U.S. Nuclear Waste Technical Review Board (NWTRB) report, "Management and Disposal of U.S. Department of Energy Spent Nuclear Fuel," December 2017, addresses DOE's spent fuel inventory which comprises a broad range of fuels, resulting primarily from defense-related activities. The report is publicly available here: https://www.nwtrb.gov/our-work/reports/management-and-disposal-of-u.s.-department-of-energy-spent-nuclear-fuel-(december-2017).

The NWTRB report does not address spent nuclear fuel from commercial power reactors, however; the NWTRB report specifically points to the significant differences between spent nuclear fuel from commercial power reactors and the DOE spent fuel inventory. For example, page 4 of the Executive Summary of the NWTRB report states:

> *Unlike commercial SNF, which basically has two types, the characteristics of DOE SNF vary widely. The inventory includes more than 10 different fuel compounds including uranium metal, thorium-uranium carbide, and thorium-uranium oxide. The range of cladding composition for DOE fuel is greater than for commercial fuel, with some compositions that can degrade during storage… DOE SNF is more damaged than commercial SNF. Compared with commercial SNF, there is also less knowledge about the present physical state of DOE SNF, including its degree of degradation, and the potential for further degradation.*

The NWTRB report clearly identifies the formation of significant amounts of hydrogen in storage is associated with "aluminum-based SNF", which are DOE spent fuels, not commercial power reactor fuels. Hydrogen generation to the degree identified in DOE spent fuel is not a credible concern for commercial power reactor spent fuel.__

3. *You cannot find or characterize cracks with a camera. At the October 2018 NRC Commission briefing, the NRC staff made it clear to the Commissioners that finding and characterizing cracks (e.g., depth, size, length, direction, etc.) cannot currently be done. Only more promises of future solutions.*

Response: The ability of remote visual inspection methods to detect cracking was reviewed and assessed in NUREG/CR-7246 (PNNL-27003), "Reliability Assessment of Remote Visual Examination," August 2018 (ADAMS Accession No. ML18228A516).

Remote visual testing (RVT) is a commonly used nondestructive examination method for inservice inspection (ISI) of reactor internals to detect cracking and gross component failures. NUREG/CR-7246 describes the results from an assessment sponsored by the U.S. Nuclear Regulatory Commission (NRC) and conducted by the Pacific Northwest National Laboratory, in cooperation with the Electric Power Research Institute, for evaluating the reliability of RVT methods currently being used for [reactor] in-vessel visual inspection:

- Crack opening displacement (COD) is the dominant factor in the reliability of crack detection using commercially applied RVT procedures, with crack length being weakly correlated with detection probability.
- RVT detection is likely heavily dependent on the contrast produced by the crack opening, with crack detection becoming less reliable as the COD decreases.

> Results indicated RVT will be challenged when cracks are located in the vicinity of surface features such as scratches or weld ripples, or close to the edge of welds where shadowing and/or the presence of weld undercuts may complicate the ability to detect the crack.

The Swedish Nuclear Power Inspectorate (SKI) (now part of the Swedish Radiation Authority [SSN]) has performed a systematic evaluation of service-induced crack characteristics (Ekstrom and Wåle 1995; Wåle 2006). Cracks are assessed according to crack type and material, with several different crack types considered in the analysis. Several flaw parameters were catalogued, including the flaw orientation, flaw size, fracture surface roughness, flaw tortuosity, flaw branching characteristics, COD, and crack tip radius. The mean COD for stress corrosion cracks (SCC) analyzed ranged from 16–30 μm.

The NRC staff have previously stated that the detection of chloride-induced stress corrosion cracks (CISCC) on canisters using RVT could be challenging and have instead relied on indications of localized corrosion such as pitting that can be reliably detected using visual testing methods (2007 ASME B&PV Code Section V, Article 1, Table A-110. Imperfection vs Type of NDE Method). Because the environmental and electrochemical conditions (i.e., temperature, chloride concentrations, redox potential) for pitting corrosion and CISCC of stainless steels are similar and chloride induced pitting corrosion on austenitic stainless steels can be initiated in the absence of an applied or residual tensile stress necessary for CISCC, pitting corrosion is frequently observed concurrently with CISCC.

Advanced RVT systems have been developed for finding and measuring the dimensions of fatigue cracks in aerospace applications such as turbine engine blades and airframes. These systems can conduct 3-dimensional measurements (length, width, and depth) and have been used in recent inspections conducted at the ISFSI at SONGS. Probability of detection (POD) curves for CISCC using these advanced systems have not been determined.

References

Ekstrom P and J Wåle. 1995. Crack Characterization for In-service Inspection Planning. SKI Report 95:70, Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

Wåle J. 2006. Crack Characterisation for In-service Inspection Planning - An Update. SKI Report 2006:24, Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

4. *Even if you could find cracks, then what? You have no way to repair them. Even Kris Singh, President of Holtec says repairing canisters isn't even feasible, even if you could find them, in the face of millions of curies of radionuclides being released. He said it will just introduce another area for cracking.*

Response: NRC licensees are required to maintain systems to comply with NRC regulations, system technical specifications, and NRC license conditions. If safety issues are identified with a spent fuel storage system, the licensee must pursue corrective actions to ensure that the spent fuel is safely stored. These actions would not necessarily involve replacement of major dry storage system components (e.g., canister or cask) or repackaging the spent fuel in a new system. Corrective actions would more likely include further assessment, inspection and in-place repairs similar to those that have been used on components in commercial nuclear power reactors such as the application of remote repair welding techniques or creating a secondary confinement boundary for the spent fuel if needed (e.g., using an overpack canister to provide containment).

Licensees initiate corrective actions after an evaluation of the potential non-compliance and the appropriate repair or mitigative actions that are needed to bring the component back into compliance. The NRC does not prescribe the corrective action that a licensee will take to re-establish compliance for a specific spent fuel canister design. The NRC evaluates whether the licensee's corrective action is effective and sufficient to maintain the intended functions of the important-to-safety structures, systems, and components, and remain compliant with the requirements in 10 CFR Part 72. Proposed repair methods require demonstration and compliance with an NRC-approved quality assurance program.

In the unlikely event that a storage canister must be unloaded, procedures for removing fuel from welded stainless-steel canisters are included in operational procedures of licensed designs. These procedures are included in the Safety Analysis Reports for the dry storage systems. As such, these procedures have been reviewed and approved by the NRC. Per the regulatory requirements in 10 CFR 72.236(h), spent fuel storage systems must be compatible with wet or dry spent fuel loading and unloading facilities. Performing such an activity should not be undertaken unless there is a specific safety need, based on indications that the canister is not performing adequately and only after evaluating other measures to remedy the circumstance with the canister along with the potential risks such activities, including opening the canister, could present.

5. *See my comments submitted to the NRC regarding NRC Draft NUREG-2224 on High Burnup Fuel Storage and Transport. It includes a link to the NWTRB December 2017 report and other technical information about explosion risks and other storage and transport risks. (NRC ADAMS Accession No. ML18269A037)*

   Response: Responses to public comments received on Draft NUREG-2224 will be addressed in a separate document. When completed, the responses to public comments on NUREG-2224 will be publicly available.

6. *See also these Sierra Club comments that address concerns about spent fuel management exemptions that are putting us at great risk at San Onofre and elsewhere.*
   *Sierra Club comments to NRC proposed rule for regulatory improvements for decommissioning power reactors, Docket NRC-2015-0070, March 2016 (*ADAMS Accession No. *ML16082A004) http://www.nrc.gov/docs/ML1608/ML16082A004.pdf.*

   Response: Many of the subjects included in Susan Corbett's letter regarding Docket ID NRC-2015-0070 Advanced Notice of Proposed Rulemaking (ANPR): Regulatory Improvements for Decommissioning Power Reactors Comments have previously been addressed by the NRC as responses to public comments including (1) inspection of dry storage systems; (2) the potential for aging effects including chloride induced stress corrosion cracking (CISCC); (3) security and protection against terrorism; (4) potential pyrophoricity of spent fuel storage systems contents such as zirconium hydrides; (5) risks of dry storage of spent nuclear fuel. Many of these issues were addressed in the published responses to public comments on NUREG-1927 Revision 1 (ADAMS Accession No. ML16125A534) and ISG-2 Revision 2 (ADAMS Accession No. ML16117A082). See the link: https://www.regulations.gov/docket?D=NRC-2015-0070

   The comments from Ms. Corbett were on the advance notice of proposed rulemaking and, while the staff did not prepare an explicit response to those comments, there is a general discussion of comment themes and NRC staff response in the final regulatory basis document (ADAMS Accession No. ML17215A010). The current NRC staff position on various topics is included in the proposed rule package which is before the Commission (ADAMS Accession No. ML18012A019).

   The staff considered all comments received and used them to develop the proposed rule

package but will not be providing detailed responses. The staff will consider comments on the proposed rule (when it is published in the Federal Register), and the staff will provide responses to significant public comments in the final rule package.

7. *Unless these thin-wall canisters are replaced with proven thick-wall transportable storage casks that have ASME N3 stamps and meet Nuclear Waste Policy Act and NWTRB storage and transport safety requirements, none of us are safe.*
These thin-wall canisters are lemons and must be recalled.

Response: The NRC has accepted the design of storage system confinement structures systems and components fabricated in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, "Rules for Construction of Nuclear Facility Components," Subsection NB, "Class 1," criteria. The NRC has accepted the design of fuel basket structures fabricated in accordance with ASME B&PV Code Section III, Subsection NG, "Core Supports" criteria. For other safety structures of storage systems, the NRC has accepted the design of these components fabricated in accordance with ASME B&PV Code Section III, Subsection NF, "Supports."

In order for dry spent fuel storage or transportation systems to be approved by the NRC, they must demonstrate their compliance with all applicable statutory and regulatory requirements. As an approved system, this would include the Holtec HI-STORM UMAX system (also known as the CoC 72-1040 system). In addition to being approved for storage under 10 CFR Part 72 of the NRC's regulations, the multipurpose canister (MPC) used in the 72-1040 system is designed to be transportable in a transportation system approved by the NRC under the regulatory requirements 10 CFR Part 71.

Concerns raised in earlier emails referenced monitored retrieval under the Nuclear Waste Policy Act (NWPA). The NWPA provisions regarding monitored retrievable storage do not apply to spent fuel storage or transportation systems generally, but instead, those provisions outline an option for a monitored retrievable storage facility to be considered by the Department of Energy for interim storage of spent fuel and high-level waste if certain conditions are met. (See NWPA, Subtitle C "Monitored Retrievable Storage", 42 U.S.C. 10161).

Concerns also referenced the Nuclear Waste Technical Review Board (NWTRB). The NWTRB, however, does not establish safety requirements for spent fuel storage and transportation systems. The NWTRB is an independent federal agency whose purpose is to perform independent technical and scientific peer review of the U.S. Department of Energy's activities related to managing and disposing of high-level radioactive waste and spent nuclear fuel.

The NRC determined that approved certificates of compliance are safe for storage and transportation of spent nuclear fuel.

8. *The Holtec and other thin-wall canisters must be replaced -- preferably before they start having major radioactive releases and hydrogen gas explosions.*
So far, all the NRC has done about this issue is allow Edison and others to hide the radiation levels from the outlet air vents of the NUHOMS canister overpacks. This is where radiation levels will be highest from through-wall cracks in aging canisters.
The NRC Region IV has refused to release outlet air vent radiation measurements from the San Onofre Areva NUHOMS canisters, even after multiple requests.

Response: As previously addressed in the response to Question 2 above, the formation of significant amounts of hydrogen in storage is associated with "aluminum-based SNF", which are DOE spent fuels, not commercial power reactor fuels. Hydrogen generation to the degree identified in DOE spent fuel is not a credible concern for commercial power reactor spent fuel.

All ISFSIs are required to comply with NRC regulations. Specific dose limits are defined in 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS," for normal operations and anticipated occurrences and 10 CFR 72.106, "Controlled area of an ISFSI or MRS," for a design basis accident. ISFSIs have multiple radiation monitors to assure compliance with NRC regulations. This is typically accomplished using multiple Thermoluminescent dosimeters (TLDs) on the ISFSI boundary fence. These TLDs are regularly monitored. The results of the monitoring program are one of many items, procedures, and operations reviewed by NRC inspectors. The NRC inspection reports are made publicly available, unless they contain classified, safeguards, or sensitive information. The NRC public web page has guidance on retrieving these inspection reports http://www.nrc.gov/waste/spent-fuel-storage/oversight.html.

Additional Information on hydrogen generation: During spent fuel loading operations, the water inside the storage system is removed. After completion, the dry storage systems are backfilled with helium, an inert gas that prevents the degradation of the contents and the internal components inside the dry storage system. The NRC staff has accepted vacuum drying methods comparable to those recommended in PNL-6365 (Knoll and Gilbert, 1987) and forced helium dehydration (Singh, 2006). The adequacy of drying is verified using specific procedures. For example, in vacuum drying adequate removal of moisture and other oxidizing gases is verified by maintaining a vacuum (typically a pressure of 3 torr) over a period of 30 minutes without vacuum pump operation. The acceptance criterion for the forced helium dehydration system is either gas temperature exiting the demoisturizer shall be less than 21°F for a minimum of 30 minutes or gas dew point exiting the multipurpose canister (MPC) shall be less than 22.9°F for a minimum of 30 minutes (ADAMS Accession No. ML16341B100). The drying criteria for both vacuum drying and forced helium dehydration assure adequate moisture removal. The containment cavity is then backfilled (pressurized above atmospheric pressure) with a highly pure, inert gas (e.g., helium) for applicable pressure and leak testing. Care is taken to preserve the purity of the cover (inert) gas and, after backfilling, the cover gas purity is verified by sampling. Helium leak testing per ANSI 14.5 (ANSI, 1997) is used to verify canister leak tightness.

An evaluation conducted by Jung et al. (2013) at the Center for Nuclear Waste Regulatory Analyses (CNWRA) has indicated that most of the hydrogen generated during dry storage occurs due to radiolysis of any residual water. Some of the hydrogen generated will be absorbed by the cladding as hydrides. The CNWRA evaluation of the potential for a flammable hydrogen gas concentration was based on assumed amounts of residual water remaining inside a spent fuel storage system after drying and backfilling with helium. This included free water either as liquid water such as water trapped in a damaged fuel rod or water vapor as well as chemisorbed water such as water in a hydrated metal oxide. The CNWRA evaluation showed that for 1 atmosphere helium backfill pressure (14.7 psi), a flammable hydrogen concentration could occur with a residual water amount of 17.4 moles (313 grams of water). With 5.5 moles of water (99 grams), no flammable hydrogen concentration was expected. When

the helium backfill pressure was increased to 5 atmospheres (73.5 psi), no flammable hydrogen concentration would occur even with a residual water amount of 55 moles of water (990 grams).

The minimum Holtec MPC-37 helium backfill pressure is 39 psi (ADAMS Accession No. ML16341B100). According to the evaluation by Jung et al. (2013), the minimum helium backfill pressure of the Holtec MPC-37 would prevent the formation of a flammable hydrogen concentration unless the residual water remaining after drying was greater than 17.4 moles (313 grams). Recent analyses of gas samples by Sandia National Laboratories (Bryan et al., 2019) show that residual water after spent fuel drying was approximately 100 g (5.55 moles) or just slightly more than the lowest amount residual water estimated in the CNWRA study which did not result in the formation of a flammable hydrogen mixture.

Because the amount of residual water remaining in a dry storage cask is low, hydrogen produced by the radiolysis of the residual water will not exceed 4%, which is the lower flammability limit for hydrogen (Airgas Inc., 2002). Therefore, hydrogen generation from residual water poses no credible risk during dry storage of spent nuclear fuel.

References

Airgas Incorporated, “Material Safety Data Sheet for Hydrogen and Liquefied Hydrogen,” Document #001026, Radnor, PA: Airgas, Inc. October 22, 2002.

ANSI N14.5, “American National Standard For Radioactive Materials — Leakage Tests on Packages for Shipment,” American National Standards Institute, Inc. New York, NY, 1997.

Knoll, R.W. and E.R. Gilbert, “Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel,” PNL 6365, Pacific Northwest Laboratory, Richland, WA, 1987.

Bryan, C.R., R.L. Jarek, C. Flores, and E. Leonard, “Analysis of Gas Samples Taken from the High Burnup Demonstration Cask,” SAND2019-2281, Sandia National Laboratories, Albuquerque, NM, February 2019.

Holtec International, “Certificate of Compliance No. 1040 Appendix A: Technical Specifications for the Hi-Storm UMAX Canister Storage System,” January 6, 2017 (ADAMS Accession No ML16341B100).

Jung, H., P. Shukla, T. Ahn, L. Tipton, K. Das, X. He, D. Basu, “Extended Storage and Transportation: Evaluation of Drying Adequacy,” Center for Nuclear Waste Regulatory Analyses, San Antonio, TX, June 2013 (NRC ADAMS Accession No. 13169A039).

Singh, K., “Forced Gas Flow Canister Dehydration,” United States Patent 7,096,600 B2. August 29, 2006.

9. *Thick-wall casks survived the Fukushima 2011 tsunami and 9.0 earthquake. Thin-wall canisters have no seismic earthquake rating when partially cracked.*

Response: Every licensee using an NRC approved dry cask storage system listed in 10 CFR 72.214 is required to perform an evaluation to show that for the conditions at the

ISFSI location, the dry cask storage system(s) selected will meet all applicable requirements. This evaluation, described in 72.212, "Conditions of general license issued under § 72.210," requires the licensee to consider, among other things, the range of natural hazards for the ISFSI location. The NRC inspects the 10 CFR 72.212 evaluation as part of ISFSI inspection activities.

The SONGS Seismic Design Basis as approved by the NRC for Units 2 and 3 at San Onofre (including the spent fuel pools) is for a seismic event with a peak ground acceleration (PGA) of 0.67g (g refers to the force of gravity). In 2010 the California Energy Commission directed SCE to evaluate seismic faults that could impact SONGS. From the results of the evaluation, it was postulated that a blind thrust fault exists beneath SONGS and that two adjacent faults could rupture simultaneously. Based on the postulated findings, SONGS decided to design the ISFSI to withstand a 1.5g PGA seismic event, which is more than twice the design basis.

Given this decision by SCE, Holtec decided to design the UMAX system, including the canisters, to resist a Regulatory Guide 1.60 ground spectra anchored at a 1.5g PGA. The NRC approved the UMAX system, and SONGS selected the UMAX system for the ISFSI. Due to the high ground water table at SONGS only the lower half of the UMAX system was buried below the ground surface. The upper half above the ground surface was encased in compacted engineered fill with sloping sides.

SONGS requested Holtec to perform a site-specific soil-structure interaction (SSI) analysis of the modified UMAX system for the site ground response spectra anchored at a 1.5g PGA. The results showed that the response of the modified UMAX system increased by slightly less than 15% at the top of the ISFSI pad. Holtec applied the 15% increase to the NRC approved UMAX system and showed that ample safety margins remained.

California building codes require buildings in the local vicinity to be designed to withstand a PGA of 0.38g. The table below compares the plants' seismic ratings with the two most recent earthquakes:

Seismic Event	Seismic Rating – Units 2/3	Seismic Rating – Dry Storage Systems	Seismic Activity Recorded Near San Onofre Plant
July 4	.67g	1.5g	.006g
July 5	.67g	1.5g	.015g

For further comparison to other large earthquakes in Southern California and recordings at San Onofre, on June 28, 1992, the Landers Earthquake (magnitude 7.3 and 90 miles away) produced a PGA of 0.038g. On Jan. 17, 1994, the Northridge Earthquake (magnitude 6.7 and 77 miles away) produced a PGA of 0.025g.

10. *The thick-wall casks at Fukushima were opened to check for damage. Something that cannot be and has never been done with the thin-wall welded canisters.*

Response: Welded stainless-steel canisters are leak tested prior to being put into service. This assures that the inert helium environment will be maintain inside the canister. The inert environment prevents degradation of the stored spent fuel and eliminates the need to inspect the fuel or the interior of the canister. However, if there is a safety need to open a welded canister, there is a procedure in the Safety Analysis Report which has been reviewed and approved by the NRC. The potential need to open and inspect any canister would be evaluated on a case-by-case basis and will consider the potential risks to workers and the public.

11. *The Fukushima aluminum alloy fuel baskets showed unexpected premature wear. These baskets are similar to the aluminum alloy baskets used in the U.S. thin-wall canisters. Japan has now banned the use of aluminum alloy baskets, switching to stainless steel baskets. The NRC has yet to address this issue, although they have known about it for years.*

Response: The Japanese Society of Mechanical Engineers (JSME) code cases for the use of aluminum alloys, borated aluminum alloys and a boron containing metal matrix composite material were never used as a basis for approval of these materials in any dry storage system approved for use in the U.S. The subsequent actions by JSME involving these code cases is irrelevant to any past NRC license approval or any current NRC licensing action.

In 2015, JSME withdrew 7 code cases involving 4 aluminum alloys 2 boron containing aluminum alloys and 1 boron carbide (B4C) containing aluminum alloy metal matrix composite (Al MMC) produced by powder metallurgy. The boron containing materials were developed for neutron absorber materials in dry storage systems.
The code cases that were originally approved in 2009 and 2013 were withdrawn for 2 reasons:

1. The code cases used acceptance criteria for mechanical property tests that were based on criteria developed for steels
2. The code cases did not adequately consider the potential for microstructural changes of the heat-treated aluminum alloys leading to reduction in strength as a result of exposure to elevated temperature

Borated aluminum alloys are used in some storage systems approved for use in the U.S. However, the borated aluminum alloys used in storage systems approved in the U.S. are not relied upon for their mechanical properties. The borated aluminum alloys are attached to welded stainless steel basket structure that provides the structural integrity of the basket. The welded stainless-steel basket used in storage systems approved in the U.S. are designed and constructed in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Subsection NG which defines the requirements for core support structures. Although the JSME code cases are not relevant for any storage system used in the U.S., ASME has required the use of both temperature and time at temperature dependent mechanical properties for heat treated aluminum alloys since 1999.

Storage systems approved for use in the U.S. have used a variety of neutron absorber materials including Boral, B4C-Al MMC, and borated aluminum alloys. None of the approved storage systems used in the U.S. rely on the mechanical properties of the borated aluminum alloy neutron absorber in the stress or stability analysis. Systems that use borated aluminum alloys include the following:

- 72-1004: 32PT, 24PTH, 32PTH, 61BT, 61BTH, 69BTH, 37PTH
- 72-1021: TN32
- 72-1027: TN68

ASME Code Case N-673, which permitted the use of a B4C-Al MMC as a neutron absorber and basket material in a dry spent fuel storage system, was not approved for use by the NRC. All ASME code cases not approved for use are documented in NRC Regulatory Guide (RG) 1.193. ASME code case N-673 was originally included in Revision 2 of RG 1.193 (October 2007; ADAMS Accession No. ML072470294) and the basis for not allowing the use of this code case is included in Table 1 of RG 1.193.

The NRC has approved Holtec's Metamic-HT for use as a neutron absorber and a structural material. Because this was a proprietary, non ASME code material, Holtec was required to supply material properties including mechanical properties as a function of temperature and after irradiation. Holtec supplemented their mechanical property information for Metamic-HT with fracture toughness testing data as a function of temperature. Holtec's analyses of the Metamic-HT basket performance uses the measured mechanical properties, including fracture toughness and the non-destructive examination acceptance criteria.

12. *Recent claims these canisters are transportable ignores the above issues.*

Response: As discussed in the response to the previous question, the concerns identified in the JSME code cases do not apply the NRC certified canister designs. Transportation and package certification is covered under the regulations in 10 CFR Part 71. The NRC has certified multiple systems for the transportation of nuclear materials including spent nuclear fuel from commercial power reactors. Prior to transportation, a dry storage canister must be evaluated to verify the canister and the contents meet the description in the transportation certificate of compliance This evaluation in addition to the licensee's loading campaign would be subject to NRC inspection. Verification of inspection system performance will follow well established practices that have been used for many years in the nuclear industry and must meet the requirements in 10 CFR 71.119.

13. *And how many people know that the New Mexico proposed consolidated interim storage system is the same defectively designed system as at San Onofre.*
And the New Mexico and Texas CIS plans, and the DOE Pilot plan (written by Holtec and others), have no hot cell or spent fuel pool to replace defective canisters.
The New Mexico and Holtec plan is to return leaking canisters back to sender. You know the "senders" have no plan in place to deal with leaking or otherwise defective canisters.
You know it is not safe to transport leaking or cracking canisters filled with high burnup fuel.

Response: The NRC has evaluated radioactive material transportation risks in NUREG-0170 (ADAMS Accession No. ML12192A283), "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes," December 1977, NUREG/CR-6672 (ADAMS Accession No. ML003698324), "Reexamination of Spent Fuel Shipment Risk Estimates," March 2000, and in NUREG-2125 (ADAMS Accession No. ML14031A323), "Spent Fuel Transportation Risk Assessment - Final Report," January 2014.

The most recent of these studies involving transportation accidents and the transportation of high-burnup fuel, NUREG-2125, showed that the risks associated with spent nuclear fuel transportation come from the direct radiation energy that the spent fuel emits, which is attenuated—but not eliminated—by the transportation casks shielding and the possibility of some quantity of radioactive material potentially released during a severe accident. The risk from the radiation emitted from the casks is a small fraction of naturally occurring background radiation, and the risk from accidental release of radioactive material is several orders of magnitude less.

14. *It's time to end this charade and tell the ugly truth to elected officials and the public before it's too late -- before these ticking time bomb "Chernobyl cans" start exploding.*

Response: The NRC is open and transparent with its public communications regarding the credible risks associated with spent fuel management. Attempting to equate the risks between postulated releases from a dry cask spent fuel canister to a postulated release from a power reactor is not credible. As passive storage systems, dry cask storage

systems do not have the inherent energy to produce a significant release of radioactive material in the highly unlikely event of a loss of containment function.

I appreciate your feedback, if you have any additional questions please contact me.

MCL

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[redacted]

September 8, 2020

Dr. David Victor, Chair
SONGS Community Engagement Panel
P.O. Box 128
San Clemente, CA 92674-0128

Dear Dr. Victor:

It is my understanding that a comment about canister repair that I made at the SONGS Community Engagement Panel (CEP) in October 2014 continues to be misconstrued by certain members of the public. I would like to clarify my comment in this letter, which I request you share with the Community Engagement Panel and others interested in the subject matter.

As I stated in October 2014, our first course of action is to prevent a canister from degradation during extended storage caused by environmental effects. Southern California Edison (SCE) has implemented several enhancements to the Multi-Purpose Canister (MPC) design and manufacturing, which go well beyond what is common in the industry. For example, SCE specified 316L stainless steel alloy for the MPCs' confinement boundary, which is known to be even more resistant to the harsh marine environment than the 304 alloy commonly used in the industry. SCE also specified *peening* of the canister welds, which establishes a compressive stress field in the welded region precluding Stress Corrosion Cracking (SCC) from taking hold. Further, the canister welds have been made with a groove geometry that reduces the level of tensile stresses in the weld mass. The above prophylactic measures against SCC are based in solid science and should be viewed as conferring significantly increased service life on the SONGS MPCs.

Now that the canisters are all in place at SONGS, safely storing spent nuclear fuel, extensive monitoring of the system is performed routinely as described in the SONGS Inspection and Maintenance Program. Because SCC is a slowly evolving phenomenon, even if, in the most unlikely event, a canister were to behave anomalously and begin to degrade, such degradation would be identified through the inspection process and mitigative actions would be taken well before an SCC nucleation site advances to a thru-wall crack. At the time of the CEP meeting, I stated that I did not believe it was practical to repair a canister in the unlikely event it has suffered localized degradation to the point of a crack. I further explained that I thought it would be difficult to isolate the indication and to perform a localized repair because the radiation field emanating from the canister would impede locating an isolated location on the surface of the canister. My specific comments regarding the number of curies inside of the canister referred to the radiation emanating through an intact canister shell, not leaking from the canister.

As I said then and continue to advocate, in the unlikely circumstance that a canister were to develop a crack, we can easily isolate that canister in a Part 71-certified cask that keeps it cool and provides

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Dr. David Victor, Chair
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a new robust confinement boundary, removing any reliance on the degraded canister. Our HI-STAR 190 licensed transportation cask has been designed to be compatible with the HI-STORM UMAX system at SONGS and therefore can be used for this purpose. Furthermore, over the past several years, there has been considerable progress in the development of remote robotic inspection and repair techniques in the industry. I understand that, recently, SCE has reported the successful testing of a remotely applied metallic overlay to mitigate a potential flaw in a canister. It is my understanding that SCE prefers this robotic approach as the remedial path for healing a degraded canister. In short, multiple competent solutions are available for a plant owner to deal with a SCC-afflicted canister which, as I have said repeatedly and publicly in the past, is not a credible event at SONGS during the Independent Spent Fuel Storage Installation's licensed life.

I assure you, Holtec is committed to ensuring the safe storage of spent nuclear fuel on the SONGS site and at the 132 other nuclear plants around the world that rely on Holtec's technologies to safely store and transport their spent nuclear fuel. I trust this letter clarifies the intent of my comments and their context. If you have any additional questions on this issue, please do not hesitate to contact me.

Sincerely,



Dr. Kris Singh, Chief Technology Officer
HOLTEC INTERNATIONAL

cc: Mr. Doug Bauder
Mr. Vince Bilovsky
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