RESPONSES TO PUBLIC COMMENTS ON DRAFT NUREG-1927, REVISION 1, "STANDARD REVIEW PLAN FOR RENEWAL OF SPECIFIC LICENSES AND CERTIFICATES OF COMPLIANCE FOR DRY STORAGE OF SPENT NUCLEAR FUEL"

Division of Spent Fuel Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

TABLE OF CONTENTS

CHAPTER 1: INTRODUCTION

On July 7, 2015, the U.S. Nuclear Regulatory Commission (NRC) staff published for public comment Draft NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" (80 FR 38780). The staff undertook the development of Revision 1 of NUREG-1927 to respond to lessons learned from safety reviews of renewal applications for specific licenses of independent spent fuel storage installations (ISFSIs) and certificates of compliance (CoCs) of dry storage systems (DSSs). Through the development of the draft Revision 1, the staff considered stakeholder input received at public meetings on renewal topics, including a public meeting on July 14–15, 2014, specifically aimed at soliciting stakeholder input on the staff's considerations for revisions to the guidance in NUREG-1927. Revision 1 contains expanded and new guidance to provide greater detail and clarity for the various elements of the staff's safety review of storage renewal applications.

The staff received 9 comment letters from various stakeholders, consisting of 225 individual comments. The commenters and their affiliations are shown in Table 1, along with the Agencywide Documents Access and Management System (ADAMS) Accession number of the individual comment letters. The public comments are located in ADAMS Package Accession No. ML15356A560.

	Name	Affiliation	ADAMS Accession No.	Submittal Date
1	Robert Einziger	NWTRB	ML15208A105	7/21/15
2	Ace Hoffman	Public	ML15216A355	7/31/15
3	Kristopher Cummings	NEI	ML15243A455	8/21/15
4	Marvin Lewis	Public	ML15243A452	8/24/15
5	Donna Gilmore	SanOnofreSafety.org	ML15243A451	8/21/15
6	Patricia Borchmann	Public	ML15243A250	8/22/15
7	Richard Morgal	Public	ML15243A249	8/25/15
8	Ray Lutz	Citizens' Oversight Projects	ML15243A248	8/22/15
9	Anonymous	Anonymous	ML15243A247	8/25/15

The staff reviewed and considered public comments, in finalizing the guidance in NUREG-1927, Revision 1. The staff also received input from the NRC's Advisory Committee on Reactor Safeguards on the staff's plans to finalize guidance. The staff developed responses to the public comments received on the draft guidance, which are presented in this document. This present document lists stakeholder comments by topical area related to the organization of NUREG-1927 and provides NRC staff responses to the comments. In reading the comments and responses, note the following:

- The comments have been grouped by topical area.
- The comment numbers are added by NRC staff for convenience and to simplify cross-referencing.
- The content of the comments is taken directly as provided by the commenters. In a few cases, minor editorial changes have been made by NRC staff, only for readability.
- The formatting in original comment letters has in some cases been altered to fit this present document. NRC staff has tried to retain formatting that is pertinent to the comments.

CHAPTER 2: RESPONSES TO PUBLIC COMMENTS ON GENERAL ISSUES

The comments on General Issues either: (1) reference the front matter of NUREG-1927, (2) reference the general guidance or a general concept in NUREG-1927 and are not specific to a particular topical area included in this comment response document, or (3) are not specific or related to the guidance in NUREG-1927.

2.1 Comments from Robert Einziger/NWTRB

Comment 2.1.1

Comment: This is a well written and understandable document. It is a vast improvement over Revision 0, and provides better guidance to the reviewers (and applicants) about what should be included in the renewal application request and how to provide that input.

Response: NRC staff notes the comment.

Comment 2.1.2

Comment: The one programmatic omission is the situation where fuel has been stored at one site for some time period where it undergoes degradation then is transported to a second site, where it will be stored under a new application. Components such as the canister, canister internals, and fuel may not be in pristine condition due to the storage and transportation. This degradation has to be accounted for when the components are put in the new storage system. For these components the new storage is a "quasi-renewal". Since this is a high probability situation in the near future it should be addressed in the document.

Response: NRC staff agrees with the comment. The staff added new language to the Introduction to discuss the situation mentioned in the comment, and the applicability of the guidance in NUREG-1927 to this situation.

Comment 2.1.3

Comment: The abstract really isn't an abstract telling why the work was done, how the work was done, and what were the results or outcome. It was more like guidance on who should use the document and that it may be revised in the future. Suggest providing a true abstract or summary so a reader can decide whether they want to delve into the document in detail.

Response: The staff agrees in part with the comment. Because this NUREG report is a standard review plan for the staff, and not a technical report, the Abstract does not describe how the work was done and what were the results or outcome. However, the staff did make changes to the Abstract to include a better summary of the report and to eliminate repetition with the Introduction.

Comment 2.1.4

Comment: P-ix L-33 - "KJ" should be "kJ". Only proper names are capitalized in units unless the capital has special meaning

Response: The staff agrees with this comment and has made the suggested edit.

Comment 2.1.5

Comment: P-2, L-9 - NUREG-1927 specifically excludes the MRS [monitored retrievable storage installation] from the guidance. Other than some statements in 10 CFR 72 with regard to financial responsibility (72.22(5)(ii), emergency plans (72.32(16)(b), which are excluded from this document, and statements related to the NWPA there is nothing in 10 CFR Part 72 that technically excludes the guidance in this document from also being applicable to an MRS. In light of the DOE preparation for an interim storage site, an MRS may be a possibility. Since the fuel going into an MRS would have been previously stored at another site the considerations in this document would be applicable especially the aging management. So that planners for an MRS might know the technical issues they may have to address, there should be a footnote indicating that the technical considerations in this document would also be applicable to an MRS.

Response: The staff agrees and disagrees in part. The NRC modified Title 10 of the *Code of Federal Regulations* (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" (http://www.nrc.gov/reading-rm/doc-collections/cfr/part072/), to include requirements for an MRS, to comply with the 1982 Nuclear Waste Policy Act (NWPA). The MRS is linked to the development of a geologic repository, and NWPA places the following limitations on the MRS (which are reflected in 10 CFR 72.44, "License Conditions"), in that construction: (1) must be approved by Congress, (2) may not begin until the NRC has authorized the construction of a repository, and (3) is prohibited during such time as the repository license is revoked by the NRC or construction of the repository ceases. Given the continued uncertainty in the National strategy for a geologic repository, an application for the initial licensing (or for a renewal) of an MRS is not currently expected. However, the guidance is applicable to the review of a renewal application submitted by a specific licensee, which may include DOE (as DOE can, and does, hold 10 CFR Part 72 specific licenses). The staff added language to the Introduction regarding the applicability of the guidance to aging management for systems stored at a second/subsequent storage location (e.g., interim consolidated storage facility). See the response to Comment 2.1.2.

2.2 Comments from Ace Hoffman

Comment 2.2.1

Comment: I would appreciate knowing if any NRC staff are actually required to read this submission in its entirety, and if so, at what expertise level are they?

Response: Yes, NRC staff reviewed and carefully considered all timely public comments received on the draft NUREG-1927, Revision 1. The staff involved in the public comment resolution and coinciding finalization of the NUREG-1927 guidance includes technical staff specifically assigned to the NUREG-1927 guidance revision, as well as independent review from other NRC technical staff, management, legal counsel, and the Advisory Committee on Reactor Safeguards (http://www.nrc.gov/about-nrc/regulatory/advisory/acrs.html). The collective expertise of the above involved staff includes advanced educational degrees and hundreds of years of experience in various scientific and engineering disciplines.

No changes were made to the guidance as a result of this comment.

Comment 2.2.2

Comment: The most dangerous "quap" on the planet is the quap that's being produced right now…

It's time for all good activists to come to a realization about so-called "interim storage" plans for used nuclear reactor core assemblies, aka "spent fuel" or "used fuel" – there's been a lot of terms for it over the years. What it is, is the most dangerous "quap" on the planet. And an intractable problem. ("Quap" is a term for radioactive waste coined by H. G. Wells more than 100 years ago.)

A news item in today's World Nuclear News (7/30/2015), the propaganda publication of the World Nuclear Society (publishing only the brightest spin on the news of the nuclear business globally) says that "soon," Holtec International plans to start the regulatory process for its proposed New Mexico "used fuel store." This is an interim fuel repository to hold the fuel until something replaces the proposed -- and currently abandoned -- Yucca Mountain permanent repository in Nevada, or until Yucca Mountain is restarted (it's hard to see how that could happen: It was leaking water, it was in a volcanic and earthquake-prone area, and in some places, parts of the ceiling and walls were degrading already, to name just a few of the many problems Yucca Mountain had when it was abandoned (for those who thought it was just "politics" that stopped Yucca Mountain)).

Around San Onofre, and probably around other closed nuclear power plant sites around the country, there is a strong push for one or more interim spent fuel waste storage facilities. Every obstacle to moving the fuel has been lifted in the local activist's minds, from highway and rail line infrastructure problems, to the will of the people around the proposed waste dump, the danger to the populations the waste will pass through, the thinness of the canisters which hold the waste, the inadequacies of the standard tests which each cask is supposed to pass…you name it, the activists have forgotten it.

And why?

Because leaving the waste around millions of people is ludicrous. In San Onofre's case, more than 10 million people live within 50 miles of the San Onofre waste dump, known as the Darrell Issa Nuclear Waste Dump by many citizens, and the San Onofre Nuclear Waste Generating Station by others. SONWGS is pronounced "SONGS" because the "waste" in the acronym was ignored for more than half a century, so the W was silent (and not even printed) the whole time the waste was being generated. Congressman Darrell Issa has been a strong supporter of the power plant, especially when, about 12 years ago, two vital decisions were being made.

One bad decision was to allow dry cask storage on site at all. The spent fuel pools were dangerously overfilled and Yucca Mountain wasn't moving forward fast enough, and eventually was stopped completely. We were assured the casks would be temporary, only until Yucca Mountain was opened.

The other bad decision was to force ratepayers to pay for four new steam generators (two per reactor), among the largest steam generators in the world, of a new design, to produce more power than before in the same space where the previous steam generators had been carefully

fit by the original design engineers. Numerous changes were hidden from the public (and the regulators) by claiming it was a “like-for-like” replacement, when in fact it wasn’t.

This second decision proved fatal to the plant and could have resulted in a meltdown when one of the steam generators failed on January 31, 2012. Extremely hazardous radioactive primary coolant was released to the secondary side through a tiny hole in one of nearly 20,000 stainless steel tubes inside the replacement steam generators (there were nearly 10,000 U-shaped tubes in each steam generator).

Subsequent inspection revealed another tube with 99% wall thickness worn away, and thousands of tubes with lesser wear and tear -- after only 11 months of operation. The public had been told these tubes would last 60 to 80 years with only “minimal” degradation. There would be leaks, but they would be few and far between (not like the old steam generators, which leaked extensively). It was claimed it would be a long time before so many of the tubes would have to be plugged (about 20%), that the plant would not have sufficient cooling capability to prevent a meltdown in the event of a “main steam line break,” which is considered a credible (meaning possible) accident scenario by the Nuclear Regulatory Commission. Only if enough tubes are available, can the reactor be properly cooled down.

But what happened on January 31, 2012 at San Onofre wasn’t just regular wear-and-tear at an accelerated pace, usually caused years later as parts start to fall apart and the tubes start to rattle more and more. This was something different. This wear was not random: Tubes were banging into each other at the U-shaped portion of the tubes at the top of the steam generator, and then those tubes would bang into the tubes next to them, which would bang into the next tube over…

Had the operators continued to run the reactor much longer (for example, if a control rod jammed or if they didn’t start the shut-down procedures soon enough), the tubes would have started to break away -- one, two…another…and another….

Primary coolant would have been spewing from the 2200 psi reactor side, to the 1200 psi steam (secondary) side, through two or more finger-sized holes. From both ends of the broken-off tube, primary coolant would be flowing so fast that a meltdown would inevitably follow, because makeup water could not be added to the system fast enough. If only one tube were to completely break off, the two broken ends of the tube would be flinging around wildly, perhaps breaking off other tubes.

The complete separation of even one tube would arguably be a “beyond design basis” accident (i.e., unthinkable, and the operators would not be trained to handle it, if it were even possible to handle). The complete “guillotining” of more than one tube would probably be catastrophic (resulting in a meltdown) and definitely would be a “beyond design basis” event.

But southern California was lucky that day. Federal regulations for how fast a leak is allowed to grow, and for how long, were apparently followed, and those regulations were tight enough (as chance would have it) that there was only a pinhole leak, not a complete breakaway of one or more tubes inside one or more steam generators.

In case you’re wondering why the regulations don’t simply call for immediate shutdown when a leak is discovered, it’s because emergency shut-downs (called “SCRAMS”) are highly damaging events and only a few dozen -- at most -- are allowed at any one reactor site over the entire life of the reactor. Normal shutdowns for refueling and periodic inspections are much more gentle.

Additionally, some leaks fill with “crud” and block themselves up almost entirely after a while. During the next outage, if they can figure out (with pressure testing) which tubes were leaking, those tubes are plugged at both ends and taken out of service.

The problem that shut down the reactor on 1/31/2012 was called Fluid Elastic Instability (FEI) and was well-known to the steam generator industry, but had (as far as we know) never happened in an operating reactor before, at least, not in America (it’s not known if it might have happened elsewhere, because reactor companies are notoriously secretive about their problems, to the detriment of the entire industry and the public).

As it turned out, the tube swinging was caused in part by excessively “dry” hot steam, so dry that it removed the thin film of water that was supposed to be coating the outside of the tubes, which made the banging of one tube into another all the more damaging. The tops of the tubes were supposed to be covered in very wet steam mixed with water, but the tubes in the new design were nearly six feet taller than the previous design before the 180 degree bend around to the downstream leg. There was less room above the tubes for mixing of the steam and water before it went into the dryers, which are cyclonic devices that spin the steam/water mixture and fling the water out and back around to cycle again, and perhaps become steam the next time through the steam generator. Too much water in the steam slows the system down (water doesn’t go around bends nearly as well as steam does), and damages the spinning turbine blades that produce electricity.

At some point the designers of the replacement steam generators miscalculated the recirculation ratio of the steam/water mixture. Approximately 3/4s of the water is supposed to go around the system each time -- i.e., not turn to steam. Instead, far less water was being recirculated. They were getting lots of steam and it was nice and dry. Profits were going to be good. The operators of San Onofre were very happy with the steam they were getting, until the leak occurred.

Nobody seemed to notice that something had to be wrong. Too much energy (steam) was being released compared to how much water was being pumped back around. And there were reports of workers hearing strange vibrations, a classic symptom of FEI. Nobody could go in the domes to check until the reactors were shut down, but the clues were there.

As it turned out, one of the reactors was already shut down at the time of the accident, but the operators didn’t think to check for wear of the new tubes in the new steam generators. Or more precisely, weren’t required to check, so they didn’t. It was instead simply down for a regular refueling outage, except they were also replacing the Reactor Pressure Vessel Head (RPVH), which was rusting out prematurely. In 2002, one RPVH in Ohio had rusted completely through to the stainless steel inner lining, prompting closer inspections and eventual replacements of RPVHs at dozens of other reactor sites. RPVHs weigh about 20,000 pounds each, with as many as a hundred or more holes drilled through them for the control rods and instrumentation. In Ohio in 2002, the stainless steel liner, about 1/8 inch thick, was already bulging outwards at the point were the “hole in the head” wear occurred. We almost lost Ohio.

At San Onofre, from late 2011 when the first unit to have its steam generators replaced shut down for its first post-replacement refueling and for an RPVH replacement, until January 2012 when the leak occurred in the other unit, the operators had plenty of time to inspect their new steam generators in the shut-down reactor but didn’t do so. Doing so would have revealed an astonishing amount of wear to the tubes.

After the leak in the other reactor, a close inspection of the suddenly-not-operating reactor revealed thousands of wear spots besides where the leak occurred. Finally the already-shut-down reactor was inspected and although no FEI damage was found to have occurred, nevertheless thousands of tubes were also severely worn in that reactor's steam generators too.

One thing was clear: The new steam generator design was seriously flawed. And, it wasn't "like-for-like" in any reasonable sense of the term. The public, and the regulators, had been hoodwinked. (Actually, the NRC probably knew they were being hoodwinked and looked the other way. After all, San Onofre's owners actually bragged about how they had avoided public scrutiny (and NRC scrutiny along with it) in a trade journal! That would have been hard to miss.)

Both reactors were permanently defueled and the site is now beginning to be decommissioned. Decommissioning means hauling away the other highly radioactive reactor components besides the spent fuel (such as the RPV and the RPVH, probably together), grinding the rest of the site up into small enough chunks to haul that away too, and washing most of the "slightly radioactive grinding dust" into the ocean. Except for the used reactor assemblies, aka, spent fuel. No one knows what to do with those in either the long term OR the short term. They will not leave the site, at least for now. The decision to allow dry cask storage on site could still prove to be catastrophic for humanity, and especially for southern California.

The Nuclear Regulatory Commission (NRC) is currently holding hearings and information sessions with "top" NRC officials (who know only one core concept: full faith in nuclear power).

These hearings cover spent fuel storage regulations for the coming era, expected to last up to 60 years -- but possibly much longer (300 years has been mentioned several times).

In other words, the NRC is concluding (as we speak) that enormous, round, welded-shut, permanently sealed (except for a pressure gauge or two) casks, made of 1/2-inch thin stainless steel, surrounded by a three to five feet of reinforced concrete, will be adequate for spent fuel storage near population centers. These storage casks (approximately 150 will be needed at San Onofre) will never be inspected or moved until and unless the license runs out, or a permanent or consolidated interim storage facility opens somewhere in the country.

Thank you Holtec and thank you New Mexico? Not so fast.

First of all: Thousands of citizens in New Mexico don't want the waste, even if they're told it's "temporary" (which they don't believe, having seen what happened to Yucca Mountain). Thousands more citizens along the roads and rail lines that will lead to the facility from the nuclear waste dumps like SONWGS don't want the waste traveling through their neighborhood.

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made in response to the comment.

NRC staff notes that the NRC does not have the authority to dictate whether nuclear energy or radioactive materials will be used in the United States. As long as nuclear energy is part of the National energy strategy and as long as radioactive materials are used in the United States, the NRC is responsible for licensing and regulating the Nation's civilian use of radioactive materials to protect public health and safety, promote the common defense and security, and protect the environment. The NRC has established regulations for the safe and secure storage of spent

fuel in 10 CFR Part 72, which includes provisions for renewal of licenses for independent spent fuel storage installations and designs of dry storage systems.

Comment 2.2.3

Comment: What these citizens fear is very real: Any leak would be a serious problem, but the worst problems would be a fire igniting the contents. Inside are highly radioactive ceramic pellets of uranium dioxide, as well as plutonium and other transuranics and fission products. Many of the fuel pellets are likely to already be damaged: Cracked like an old piece of china. In any case, the ceramic pellets are highly embrittled, with chipped-off bits, and radioactive dust particles that have flaked off, which would burn very easily if brought to the right temperature by some external heat source or by a criticality event (described in more detail below).

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made in response to the comment.

The staff understands and values the concerns on potential flammability/combustion of the spent fuel stored in dry storage systems. To address this concern, the staff reviews the design bases of each dry storage system and ISFSI to ensure that the fuel will be reasonably maintained in an analyzed configuration and flammability/combustion of the spent fuel is not a credible event during normal and off-normal conditions of storage. The staff's review ensures these conditions are met by the following defense-in-depth measures: (1) materials used in the dry storage system remain stable (i.e., materials do not inadvertently lead to the generation of combustible/flammable gases (e.g., radiolysis in confined spaces)), (2) adequate drying processes are used for limiting the presence of oxidizers/flammable gases in the confinement cavity, (3) overpressurizing of the canister/cask with a highly pure inert gas to mitigate the entry of oxidizing gases into the canister/cask, (4) cladding integrity or canning of grossly breached spent fuel to maintain a known fuel configuration, and (5) daily monitoring for ensuring that decay heat is adequately removed from the cask/canister (overpack inlet/outlet vent inspections, temperature monitoring) and the systems do not reach temperatures that may compromise cladding integrity. The staff notes that the structural integrity of the fuel pellet is not relied upon for ensuring the dry storage system can perform its intended safety functions. Additional details on how the staff conducts reviews to verify the above conditions are discussed in NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility." Under the requirements of 10 CFR 72.42(a) and 72.240(b), licensees requesting renewal are expected to have aging management programs to ensure that the spent fuel remains in the analyzed configuration and the confinement barrier (canister/cask) remains in the analyzed configuration, thereby ensuring that the spent fuel remains subcritical during the period of extended operation.

Comment 2.2.4

Comment: Gamma radiation is the immediate danger for anyone working with spent nuclear fuel. It will take many generations for that not to be a problem. Gamma rays get through the 1/2-inch thin stainless steel cask with little reduction in strength. That's why there is also three to five feet of concrete overpack surrounding the dry fuel canisters.

During transport, there is an additional overpack, but it is not as thick as the concrete overpack. Instead, there are special rules about how long the fuel can remain in one place on the highway, such as when a driver stops for dinner at the same diner everyone else (including babies and pregnant women) is also stopping at.

It is utter fallacy to think the rules for safe transport of nuclear waste are safe enough, especially concerning exposures to woman (more than men), to children, to infants, and most of all, to the fetus of a pregnant woman, and to the eggs that fetus could be carrying.

Hopefully some day, the absolute risk to the most at-risk segment of our society will determine legal dose rates for radioactive substances. But for now, the rules are based on the least-sensitive person to radiation: An adult male in good health to begin with.

Ludicrous?

Everything about the nuclear business is ludicrous!

Holtec plans to open their Interim nuclear waste storage facility in 2020. That's certainly optimistic, but it helps sell the idea to keep the date in the near future. The Yucca Mountain plan was promoted for nearly 30 years and was never more than ten years in the future at any one time! And not once would any NRC or utility spokesperson at any public hearing respond to the request to stop making unmanageable nuclear waste with anything other than "Yucca Mountain!" for the entire 30 years!

Yet here we are, and Holtec is screaming "New Mexico will take it!"

I don't think so, but if it does happen, it will be a disaster for everyone, because it will falsely enable the entire industry to claim there is a "solution" to the nuclear waste problem.

But moving spent fuel nuclear waste from one place to another does little for the general population in the case of a spent fuel nuclear fire, criticality event, or cladding fire (three separate types of events which I'll go into in detail below). The radioactive debris from such events would travel hundreds of miles in the first few hours, and spread globally for thousands of generations.

Catastrophic events can occur during transport, through accidents which are known to happen from time to time on the nations roads and rail lines: Bridges collapse, sometimes onto other roads or rail lines. There have been unreachable train fires in tunnels which lasted for days. The spent fuel canisters are only expected to last about 20 minutes in a typical fire. No dry cask storage system anywhere in the world is capable of withstanding a jumbo jet impact, although probably the German ductile steel dry casks, which are nearly 20 inches thick, are orders-of-magnitude safer than the 1/2 inch thin American/French stainless steel dry cask system. But none are adequate to resist a jumbo jet and ensuing fire.

So even if Holtec's spent fuel facility is built in New Mexico, and even if it is only made available to "closed" facilities, it will be a disaster globally not just because of the risk of a transport accident as fuel is moved to the site, and not just because a jumbo jet could crash into the site in New Mexico, but also because having such a facility will enable dozens of reactor sites to remain open. They'll be able to claim that some day their fuel will either go to the Holtec site, or to some other "interim facility" (mind the gap between an "interim" facility and a "permanent" repository).

If the site does open, transporting the spent fuel there requires several dangerous (risky) steps:

First: Opening the concrete door at the front of the overpack (depending on design) while the workers remain shielded from the gamma radiation that will be emitted.

Second: Pushing a thick overpack (larger than a school bus!) up to the opening. The overpack has lead, boron, polystyrene and/or other special radiation-absorbing elements and lots more stainless steel. The transport overpack is expected to be used over and over.

Third: After the fuel canister is moved into the transport overpack, the combined assembly is moved slightly away from the cement overpack, where the canister sat, unopened, for perhaps 60 years or more, and then a rear seal is put in place on the back end of the transport overpack.

The longer we wait to make this transfer, the lower the radiation doses received by the workers—or by anyone near an accident. Maybe more of the procedure can be automated, too.

Once the cask is out of the concrete overpack, its journey has only just begun. By rail or by truck? This hasn't been decided, but the general feeling is it should go by rail, because rail lines tend to only go through less populated areas (pure coincidence that they are also usually the poorest communities), and they can carry more weight. That means thicker shielding can be used. Or bigger dry casks, which is what the industry supports (32 or more fuel assemblies, instead of the previous standard of about 24).

San Onofre shut down suddenly, and did so, thankfully, without melting down the reactor(s) and without causing the loss of some of the most expensive real estate in the world. When the decision was announced to permanently close the plant, suddenly the local activists could completely focus on the waste problem that had been silent for so long. And of course, all of them (myself included) want the spent fuel to be moved away. But to where? And whose rights will we have to violate to get the waste there? And what will moving it do to the perception that other nuclear power plants also have a solution to the waste problem because we managed to somehow get rid of ours?

One of the local activists, who desperately wants the waste moved, said he is willing to overlook every obstacle: The trains would move more slowly than other trains, so the casks would be less likely to fall off the tracks or be cracked open in an accident. Well, absolutely, those are benefits of the train or truck traveling relatively slowly. But by going slowly, they will also be under and on bridges longer, and they will be easier targets for an ambush by terrorists.

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made in response to the comment.

The NRC established safety requirements for transportation of radioactive material in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material" (http://www.nrc.gov/reading-rm/doc-collections/cfr/part071/), and related information may be found at http://www.nrc.gov/waste/spent-fuel-transp.html. The NRC established safety standards for protection against radiation, including public dose limits, in 10 CFR Part 20, "Standards for Protection against Radiation" (http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/), and related information may be found at http://www.nrc.gov/about-nrc/radiation.html. The regulations in 10 CFR Part 72 also include dose limits for spent fuel storage. The current requirements in 10 CFR Parts 20 and 71 are protective of public health and safety and the environment.

Comment 2.2.5

Comment: Nobody should forget about terrorists when thinking about what to do with nuclear waste. These days, enormous acts of violence impacting hundreds or even thousands of lives are being committed by seemingly-sane people, such as pilots and copilots of jumbo jet aircraft. GermanWings. MH370. But those are only the most recent examples. Plane hijackings (and crazed pilots) have been occurring for decades. One hijacked airplane in the 1970s overflew very close to Oak Ridge, Tennessee (the hijacker went on to Cuba, where he was promptly arrested). One military pilot hijacked a A-10 Warthog, and flew it into a mountain. His Depleted Uranium shells could easily have pierced any American/French dry cask storage system, and probably the German casks, as well.

Someone may be sick enough to aim for a dry cask some day. The best protection is to separate the fuel out into as small a bundle as possible (to reduce the potential for a criticality event) and place that package within enormous earth berms and a thick concrete overpack—but yet, the containers have to be able to be inspected.

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made in response to the comment.

The NRC's mission is to license and regulate the Nation's civilian use of radioactive materials, to protect public health and safety, promote the common defense and security, and protect the environment. In terms of spent fuel storage, the NRC's strategy for protecting public health and safety, the common defense and security, and the environment focuses on ensuring that its requirements, in combination with the design features of dry storage systems and ISFSIs and licensees' security measures, are effective in protecting against the potential effects of terrorist attacks on ISFSIs.

The NRC provides security requirements for physical protection for spent fuel storage and transportation in 10 CFR Part 73, "Physical Protection of Plants and Materials" (http://www.nrc.gov/reading-rm/doc-collections/cfr/part073/), 10 CFR Part 72, and orders that provide additional security measures. These security requirements provide high assurance that terrorist attacks cannot endanger the public's health and safety by intentionally releasing radiation from an ISFSI. The NRC reviews and approves facility security plans in evaluating the adequacy of on-site security measures. The NRC also inspects ISFSIs to ensure licensees' complete and correct implementation of the features of the security plan as well as the applicable regulations and orders.

The NRC has initiated several actions designed to provide high assurance that a terrorist attack would not lead to a significant radiological event at an ISFSI. These include: (1) the continual evaluation of the threat environment by the NRC, in coordination with the intelligence and law enforcement communities, which provides, in part, the basis for the protective measures currently required, (2) the protective measures that are in place to reduce the chance of an attack that leads to a significant release of radiation, (3) the robust design of storage casks, which provides substantial resistance to penetration, and (4) NRC security assessments of the potential consequences of terrorist attacks against ISFSIs, that inform the decisions made regarding the types and level of protective measures. Over the past 20 years, there have been no known or suspected attempts to sabotage, or to steal, radioactive material from storage casks at ISFSIs, or to directly attack an ISFSI. Nevertheless, the NRC is continually evaluating the threat environment, to determine whether any specific threat to ISFSIs exists.

After issuance of security orders for ISFSIs in 2002, the NRC used a security assessment framework as a screening and assessment tool to determine whether additional security measures, beyond those required by regulation and the security orders, were warranted for NRC-regulated facilities, including ISFSIs. In conducting the security assessments for ISFSIs, the NRC chose several storage cask designs that were representative of most currently NRC-certified designs. Plausible threat scenarios considered in the generic security assessments for ISFSIs included a large aircraft impact similar in magnitude to the attacks of September 11, 2001, and ground assaults using expanded adversary characteristics consistent with the design basis threat for radiological sabotage for nuclear power plants. The resulting generic assessments formed the basis for the NRC's conclusion that there was no need for further security measures at ISFSIs beyond those currently required by regulation and imposed by orders issued after September 11, 2001.

Additional information may be found at http://www.nrc.gov/security.html.

Comment 2.2.6

Comment: …but yet, the containers have to be able to be inspected. They have to be retrievable. You can't just bury the stuff and forget about it. You have to be able to deal with "problems" such as:

Rust.

Embrittlement, stress corrosion cracking (SCC), chloride-induced SCC, Wigner's disease, hardening of the arteries, aging…it's all basically the same thing, and happens to the stainless steel cask, the zirconium fuel rods, the fuel pellets inside the zirconium fuel rods, the steel assemblies holding the fuel rods in place, the steel panels which keep each fuel assembly separated from each other inside the cask (to prevent a criticality event)…every part. They'll all embrittle.

After 60 years, everything will be much more fragile than when it was removed from the reactor. The very fact that they will be a lot cooler temperature than they are now means they will be more brittle. It's even been proposed that "if necessary" the casks could be heated for the duration of the transfer to the "interim" storage facility -- though how they'll know which ones would need heating has not been determined, let alone how to be sure they are heated enough, but not so much that they can sag and break if put under too much stress.

Response: The regulations in 10 CFR Part 72 require renewal applicants to include a description of their aging management programs for management of aging issues that could adversely affect structures, systems, and components (SSCs) important to safety, and NUREG-1927 provides guidance for review of an applicant's aging management programs (AMPs). The applicant's aging management review considers the SSCs, the materials of construction, the service environment, and the applicable aging mechanisms and effects. The aging management review includes aging mechanisms and effects that have actually occurred, as well as those that could be reasonably expected to occur. This is already discussed in Section 3.4 of the guidance. Therefore, no changes were made to the guidance.

Comment 2.2.7

Comment: Moving spent fuel safely is not easy. In fact, it's impossible to be sure it can be done safely. That's why the industry resorts to Probabilistic Risk Assessments (PRAs) which assume that accidents might happen, they just won't happen very often.

But with 10,000 dry casks needed for all the fuel that's been produced so far (and more than 2,000 dry casks in use at the moment), there is plenty of room for "rare" accidents. If something has a rarity of one in a million for one transfer of one dry cask, then the chance of it occurring is 1% if you are moving 10,000 casks. Nuclear waste accidents are far too severe to be allowed to happen that often. The industry claimed reactor meltdowns had less than a one in ten thousand year chance of occurrence. So far, there have been four meltdowns in just over half a century of nuclear operation. PRAs are fanciful, complicated, and misleading.

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made in response to the comment.

The NRC establishes safety requirements for transportation of radioactive material in 10 CFR Part 71 and believes the current requirements are protective of public health and safety and the environment. See the response to Comment 2.2.4. As the comment refers to risk assessment, additional information may be found at http://www.nrc.gov/about-nrc/regulatory/risk-informed.html.

Comment 2.2.8

Comment: What could possibly go wrong?

A cladding fire would be very bad -- it's what happened at three reactors in Fukushima. The zirconium cladding surrounding the fuel pellets is pyrophoric (ignites easily in air). In Fukushima, as the water boiled away, the cladding was exposed to steam. When that happened, the zirconium split the hydrogen atom in the steam molecules away from the oxygen atoms. The hydrogen accumulated and then exploded. That happened at least twice at Fukushima and possibly three times, and nearly happened at Three Mile Island. At TMI, a large hydrogen "bubble" accumulated at the top of the reactor containment dome but did not explode. Fukushima's big boxes accumulated the hydrogen outside the reactors' "containment vessels," which are much smaller than the containment dome at Three Mile Island.

At Fukushima the accumulated hydrogen exploded, but one of the reactors apparently accumulated the hydrogen in a different building. There were apparently ventilation interconnects between the reactor buildings, and the released hydrogen from one of the reactors transferred over to a different reactor building before exploding. At least, that what TEPCO, the owners, claims happened. (It should be noted that TEPCO is perhaps the most dishonest company on the planet, although every nuclear corporation is a contender.)

A spent nuclear fuel fire would include a cladding fire but would be much worse. It is when the ceramic uranium dioxide fuel pellets themselves are burning. Fukushima, TMI and Chernobyl all experienced fuel melts, but not burning fuel pellets (as far as we know). Chernobyl had some fuel exploded into pieces or "chunks," as well as the melted fuel. The melted fuel at Chernobyl is known as an "elephant's foot" and has been photographed, but has been left to continue to smolder (i.e., produce fission products which are released). A new containment

structure was recently built, but another and another will eventually be needed, for thousands of years, each one more expensive than the last.

If a criticality event occurs with spent fuel, the area around the site will be permanently contaminated. That area could stretch for hundreds of square miles downwind of the accident, with many other areas further away, or not predominately downwind, nearly as contaminated.

The "spark" needed to touch off a fuel pellet fire has to be very hot and/or last a long time -- but a terrorist would know this and plan accordingly, if that's what they wanted to do, and a jumbo jet accident could lead to a spent nuclear fuel fire if the fuel burns around the canisters long enough, or if the canisters have been breached. There would not be any possibility of fire fighters putting out the blaze: Unprotected spent fuel kills in seconds.

But besides that, putting water on a spent fuel fire might result in a criticality event, because water is a "moderator" which slows down neutrons. Slow neutrons can split additional uranium and plutonium atoms (fast neutrons, i.e., those not slowed down by a moderator, are incapable of sustaining a chain reaction in spent fuel). A criticality event could occur simply if the breach of the containment happens when its raining.

A lot can go wrong with spent nuclear fuel assemblies, and many of these dangers last for thousands of human generations. After roughly six centuries, the danger from a cladding fire is reduced to a tiny fraction of what it is currently, because the fission products that could be released today will have almost all decayed away after 600 years. Even after 60 years, the danger from fission product releases will be significantly reduced, but the plutonium and uranium risks will barely have changed.

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made in response to the comment.

The regulations in 10 CFR Part 72 provide safety requirements for storage of spent fuel, including confinement of radioactive material, radiation shielding, and sub-criticality control, under normal and off-normal operating and accident conditions. Additional details on how the staff conducts safety reviews may be found in NUREG-1536 (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1536/r1/) and NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities" (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1567/).

Comment 2.2.9

Comment: The reduction in fission product inventory in spent fuel is probably the best argument for leaving the waste where it is for now, even though it's in an earthquake zone, a tsunami inundation zone, a large and growing population center, and just a few hundred feet from a major highway where more than 100,000 vehicles pass every day -- a terrorist's delight to have such close access.

But whatever is done with the waste, it had better not enable other nuclear power plants to remain open.

15 or so years ago, when dry casks were first being contemplated to "temporarily" store nuclear waste at San Onofre and other nuclear power sites across America, many activists endorsed dry cask storage. Surely those not around then must be asking why those activists didn't

demand San Onofre shut down permanently instead of allowing an endless number of dry casks to be built and stored on site. The answer was that they believed dry cask storage was “safer” than overcrowded spent fuel pools.

And taken out of context, they were probably right. But that sort of thinking enabled San Onofre to replace its steam generators and try to keep operating for perhaps 60 more years -- that was the plan -- had the steam generators not failed that day in January, 2012, and caused the permanent shut-down of the facility.

The “interim” storage facility planned for New Mexico has no exit strategy -- and yet might become the first of several “interim” storage sites across the country.

Before any interim facility opens, Holtec plans to build for San Onofre -- if they get the contract, which seems almost inevitable at this point -- a “temporary” cement entombment system consisting of the 1/2 inch thin stainless steel canisters, placed vertically in several cement “islands” at the closed SONWGS site. Each dry cask will be placed in a separate hole in the cement with three to five feet of concrete between each cask. Adequately inspecting the casks after installation will be impossible.

At some point, the canisters will have to be pulled up into a temporary overpack, then the overpack will have to be tilted to the horizontal position in order to be placed on a transport vehicle. Catastrophic errors can occur at any stage of the operation.

Activists were wrong when they endorsed dry cask storage. They should have demanded permanent shutdown instead.

Activists now endorsing an interim storage facility are wrong as well. They should demand permanent shutdown instead. No one, no country, can sensibly decide what to do with nuclear waste until nuclear power has been abandoned permanently forever everywhere. Nuclear power -- and nuclear waste -- is not compatible with human life. It is not compatible with a small enclosed ecosystem (and if you still think the world is vast, remember you can circle it in 90 minutes from space and in about eight hours in a supersonic jet (not counting slowing down to refuel)).

And speaking of air transport, earth-bound vacuum tunnels are far safer not only for the passengers (they need no crew) but for everything that would otherwise be under the airline routes. Ukraine has nuclear power plants. MH17 could have crashed into one of them when it was shot down. The GermanWings copilot could have crashed into numerous reactors -- and no one could have stopped him.

This is not the world we want to leave our children. Solar and other renewables can replace all the coal, oil, nuclear and gas power sources. There is no reason for the madness of nuclear power to continue. Finding a safe place for the waste -- and getting it there -- is impossible. Permanent shut-down of all nuclear facilities is the most important thing to do right now. Stop making more waste because there will never be a safe solution.

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made in response to the comment. See the response to Comment 2.2.2.

2.3 Comments from Nuclear Energy Institute

Comment 2.3.1

Comment: 1, 21. Delete "revised technical." Not needed. Could be new rather than revised.

Response: The NRC staff agrees and made the recommended change.

Comment 2.3.2

Comment: 1, 40. Change "developed" to "approved." Only should be using approved ISGs.

Response: The NRC staff agrees with the comment. The staff changed "developed" to "issued" to denote that this is approved final guidance.

Comment 2.3.3

Comment: XX, Appendix B. It would be beneficial for one of the example AMPs to have an example of a TLAA/AMP for a general license with different sites. A site on the coast will experience different environmental conditions than a site in the desert. This will likely result in different degradation mechanisms, TLAAs and AMPs. An example of how the same cask can have a different AMP or TLAA at each of two sites would be helpful.

Response: The staff considers additional AMP guidance to address additional storage scenarios to be outside the scope of NUREG-1927. The Managing Aging Processes in Storage Report, which is currently in development, will contain more detailed AMP guidance, and this comment will be taken under consideration for that document. The staff concluded that no change was necessary.

Comment 2.3.4

Comment: XX, Appendix B. The level of detail contained in the example aging management programs would be inappropriate for a License Condition or inclusion in the Tech Spec, however would be more appropriately addressed in the implementing procedures. Reliance should be on the existing Corrective Action Program and NRC inspectors consistent with what is required for operating reactors that have undergone license renewal.

Response: The staff's conclusion regarding including AMP information in the CoC is addressed in the response to Comment 3.2.1.

2.4 Comment from Marvin Lewis

Comment: Please allow me to join in Donna Gimore's comments to show my agreement with her observations and conclusions.

Response: NRC staff notes the comment. Please refer to comments in Sections 2.5, 5.3, 8.3, 9.3, and 10.3 for NRC staff responses to Ms. Gilmore's comments.

2.5 Comment from Donna Gilmore/SanOnofreSafety.org

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 below. 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-caskstoragedgilmore2014nov19.pdf

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 mentioned below. 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

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made to the guidance in response to the comment. However, NRC staff disagrees with some of the assertions in the comment.

The revised guidance in the draft NUREG-1927, Revision 1, provides additional guidance on the aging management review and the elements of an effective aging management program, which is greatly expanded from the guidance in Revision 0. For renewal of the existing ISFSIs and DSSs that have been licensed and certified, the NRC requires that renewal applications include time-limited aging analysis (TLAAs), if applicable, or aging management programs for management of issues associated with aging that could adversely affect SSCs important to safety. The applicant must demonstrate that aging issues or effects will be managed, so that SSCs continue to meet their intended functions, and that the ISFSI or DSS continues to perform as designed and meet the safety requirements in 10 CFR Part 72.

Regarding thicknesses of components or variations in design of dry storage systems, the NRC does not dictate specific design details or use of specific materials. Rather, the NRC sets performance-based safety requirements, and an applicant must demonstrate how its proposed design will meet these requirements in the original license application. The NRC applies the same safety standards to all storage systems. The staff also notes that there are pros and cons to each material and its properties, and the materials of construction may be selected by applicants based on the totality of their properties and performance during the storage period, which support the safety functions of confinement of radioactive material, radiation shielding, sub-criticality control, heat-removal capability, structural integrity, and retrievability. The NRC's materials review considers not only the design and material dimensions, but also the materials' physical, chemical, and mechanical properties to determine whether: materials are compatible with site characteristics, service environments, and environmental conditions; and materials performance will support the safety functions for the DSS or ISFSI during normal and off-normal operating and accident conditions.

2.6 Comments from Patricia Borchmann

Comment 2.6.1

Comment: The Description above indicates there is a "time limit not to exceed 40 years", to apply to Renewals of ISFSI Special Licenses, and COCs for Storage Cask Designs. By itself, that limited description is deceptive, and less than full disclosure. Stakeholders feel the Final revision for NUREG 1927 (Rev. 2) needs to include additional discussion to confirm how this project relates to NRC action taken August 26, 2014 ; to allow the potential indefinite continued storage of spent fuel onsite at nuclear power plants in United States, including San Onofre (SONGS 2 & 3).

When NRC Commission took that action last year to allow 'indefinitely prolonged, continued onsite storage of spent nuclear fuel", a critical new variable was introduced, that still needs to be explicitly reflected in the next draft for Revised NUREG 1927 (Rev. 2). Until an expanded description is undertaken for the Final Revision for NUREG 1927, this document is grossly incomplete.

Allow me to remind NRC staff that this is not a new request from stakeholders. In earlier written comments by D. Gilmore from sanonofresafety.org (dated 12 22 14), Gilmore already informed NRC that the Commission's extended storage decision recognized there may not be a geological repository for forseeable future, and the Licensee for San Onofre (Southern California Edison) ignores the fact, and instead relied on unsupported expectation that Department of Energy (DOE) will being to pick up spent fuel from San Onofre in 2024. It is readily foreseeable, and already known to be an unrealistic, and probably infeasible timeframe expectation. Gilmore's earlier comments confirmed SCE's PSDAR, Dcommissioning Cost Estimate (DCE), and Irradiated Fuel Management Plan (IFMP) all required the same correction.

For purpose of full disclosure and consistency, stakeholders feel it is imperative that NRC define how both short and longer term impacts will be analyzed, mitigation measures, preventive measures, and corrective actions will be applied to minimize potential impacts on public health and safety during the maximum onsite storage of spent fuel, (including High Burnup Fuel) during both short term (up to 60 years); and long term (up to 160 + years and possibly for centuries.

Response: The comment indicates that it is directed toward the "final revision for NUREG-1927 (Rev. 2)." However, the staff is interpreting that this comment is directed toward the current draft Revision 1 that was published for public comment and will consider it in that context. This comment highlights the important distinction between the NRC staff's environmental and safety reviews.

NUREG-1927 is focused on the staff's *safety* review of spent fuel storage renewal applications (as is indicated in the Abstract and Introduction to the guidance), per 10 CFR Part 72. NUREG-1927 is not necessarily limited to the first renewal period, and it can be applied to subsequent renewal periods. NUREG-1927, along with other future guidance on renewals and aging management, is expected to be updated over time to include new information, knowledge, and experience regarding aging management considerations. In addition, the "learning" aspect of AMPs (as presented in Section 3.6.1.10 of the guidance) will consider and respond to operating experience and information over time. In this way, the guidance and the framework is not a static product, but is dynamic and will change as needed. The framework is "learning," in that it will continue to be informed by future research and operating experience, as ISFSIs and dry storage systems enter the period of extended operation.

The staff conducts a separate *environmental* review to comply with the National Environmental Policy Act (NEPA), and the NRC's requirements for NEPA compliance/implementation are in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions" (http://www.nrc.gov/reading-rm/doc-collections/cfr/part051/). Regarding the commenter's reference to the 2014 NRC action on continued storage, this action was to generically determine the environmental impacts of continued storage of spent fuel, as is presented in NUREG-2157, "Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel" (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2157/). Because the timing of repository availability is uncertain, the generic environmental impact statement in NUREG-2157 analyzes potential environmental impacts over different timeframes, including an indefinite timeframe to address the possibility that a repository never becomes available.

NRC staff conducts both a safety review and an environmental review of a storage renewal application. The environmental review may reference the generic analysis provided in NUREG-2157.

No changes were made to the guidance as a result of this comment.

Comment 2.6.2

Comment: Under the Appendix B NUREG 1927 (Rev.1) text currently proposes "to use achievable, actionable acceptance criteria". Stakeholders specifically request, and deserve expanded acceptance criteria to also reflect measurable (quantitative) criteria that is capable of verification, and is evidence-based.

Response: The staff considers the current guidance in Section 3.6.1.4 and Appendix B that the inspection method/technique is qualified to meet the stated quantitative criteria, to sufficiently address the commenter's concern. Also, the staff notes that 10 CFR 72.158, "Control of Special Processes," requires that applicant's ensure that special processes, like nondestructive examination, be "controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements." No changes were made to the guidance as a result of this comment.

Comment 2.6.3

Comment: Under current Appendix B, text "proposes use of consensus codes and standards where practicable for examination methods, equipment, calibration, acceptance criteria, and personnel qualifications". Stakeholders believe this by itself, is insufficient because it is overly vague, and fails to identify a specific safety-based standard which should clearly be a proven capability (and not just a SCC intention-based function), because it allows Licensee an excess of latitude – "where practicable". Exceptions to public health and safety should not be based on practicality for Licensee convenience, or profit margin. Stakeholders specifically request, and deserve the explicit statement of mandated safety standard performance capability (not just an unproven assertion, or unverifiable claim).

Response: The staff disagrees with the comment and did not make a change to the guidance as a result of this comment. At this time, the inspection of dry storage systems does not fall under existing codes and standards (although there is current work to develop such standards). However, there are existing standards for other applications (e.g., power plant components) that, in part, may be effective for the inspection of dry storage systems, and these have been referenced in two of the example AMPs (American Concrete Institute (ACI) standards for concrete and American Society of Mechanical Engineers (ASME) standards for welded stainless steel canisters). If the applicant proposes other standards, the staff will evaluate them on a case-by-case basis to ensure that the inspections are capable of identifying degradation prior to a loss of function.

Comment 2.6.4

Comment: Under current Appendix B, NUREG text proposes to "rely on Licensee Quality Assurance and Corrective Action Programs for Further Evaluation, characterization, and other actions needed to preserve the SSC-intended functions". Based on performance at San Onfore SONGS 2 & 3 by Licensee SCE, and performance at Diablo Canyon NPP by Liensee PG&E, stakeholders in California have zero reason to have confidence in Licensee capabilities, or reliance on License Quality Assurance and Corrective Action Programs for Further Evaluation. This lack of confidence by stakeholders is based on series of catastrophic mistakes and actions taken by Licensee SCE at SONGS, and the recent June 7, 2015 NRC Event Report documented that "two spent fuel casks had been loaded improperly at Diablo Canyon in Avila Beach, CA. Upon further inspection, it was discovered that 19 of the 34 dry casks that have been loaded at the Independent Spent Fuel Storage Installation (ISFSI) have been loaded improperly." (Sierra Club, Grassrootsnetwork/team-news)

Response: The staff disagrees with the comment and did not make a change to the guidance as a result of this comment. As stated in the NRC's fact sheet on "Oversight of Nuclear Power Plants," (ADAMS Accession No. ML060690104), licensees have direct responsibility for operating their facilities safely, and the NRC's role is to provide oversight through its inspection program. The NRC is not in a position to make all safety decisions for a licensee, nor is that the NRC's role. The NRC conducts inspections to examine whether licensees are performing activities in accordance with radiation safety requirements, licensing and certificate of compliance requirements, and quality assurance program commitments. When a safety problem or failure to comply with requirements is discovered, the NRC requires prompt corrective action by the licensee that is reinforced, if necessary, with enforcement action. Additional information on the NRC's oversight program for spent fuel storage may be found at: http://www.nrc.gov/waste/spent-fuel-storage/oversight.html. Also, regarding the example

provided in the comment, when misloads are identified, licensees must conduct an evaluation, and in no case has there been a resulting safety concern.

Comment 2.6.5

Comment: At the Seabrook NPP, where ACR concrete degradation was recently identified this summer, NRC issued two green non-cited violations. The first deficiency finding indicated Seabrook Station "was not aggressively implementing" its structural monitoring program and therefore did not identify the degradation in the Containment Enclosure Building (CEB); and the second finding indicated Licensee NextEra did not provide an adequate "prompt operability determination", which would detail how the plant would address the issue in the future. Since Seabrook currently has a pending license renewal application, Neil Sheehan (NRC spokeperson) said the decision on whether to grant the plant's license renewal is heavily dependent on NextEra's "long term' plan to address the ASR (alkali-silica reaction).

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made to the guidance in response to the comment.

The staff notes that although, to date, no operating experience on alkali-silica reaction (ASR) has been reported in any concrete structure used in dry storage systems, the staff recognizes the potential for ASR during periods of extended operation for the reasons identified in Information Notice (IN) 2011-20 (ADAMS Accession No. ML112241029). Therefore, the example AMP for reinforced concrete structures included in Appendix B assumes ASR as a potential operable degradation mode and includes activities for ensuring that ASR, if it occurs, is timely detected and addressed to ensure no loss of intended function for the concrete structure.

In addition, as the comment is related to concrete degradation at Seabrook, additional information on this topic may be found at:
http://www.nrc.gov/reactors/operating/ops-experience/concrete-degradation.html.

Comment 2.6.6

Comment: Under Wet Storage Issues (page 9), authors of a 2011 NRC-sponsored study indicated that accurate assessment of again of spent fuel pools is uncertain because "it is often hard to assess their in situ condition because of accessibility problems….Similarly a portion of the listed concrete structures are either buries or form part of other structures or buildings, or their external surfaces are invisible because they are covered with liners." 39

In last paragraph on page 9, Alvarez describes how "High-density racks in spent fuel pool in U.S. power plants post potential criticality safety concerns associated with deterioration of neutron absorbing panels that allow spent fuel rods to be more closely packed. Since 1983, several incidents occurred at reactors around the U.D. with these panels in which the neutron absorbing materials deteriorated, and in some causes bulged, causing spent fuel assemblies, containing dozens of rods each, to become stuck in submerged storage racks in the pools. The problem could lead to structural failures in the storage racks holding the spent fuel rods in place."

At top of page 10, Alvarez memo indicates: "According to the NRC in May 2010";

The conservatism/margins in spent fuel (SFP) criticality analyses have been decreasing….The new rack designs rely heavily on permanently installed neutron absorbers to maintain criticality

requirements Unfortunately, virtually every permanently installed neutron absorber, for which a history can be established, has exhibited some degradation. Some have lost a significant portion of their neutron absorbing capability. In some cases, degradation is so extensive the permanently installed neutron absorber can no longer be credited in the criticality analysis (emphasis added). 40

For example, in 2007, SCE reported to NRC that Boraflex neutron absorbing panels have deteriorated to the point at SONGS 2 & 3 SFP where it was doubtful they could be credited to prevent criticality, SCE proposed installing borated stainless steel tube guide inserts, and to add more neutron absorbing boron to the pool water. 41 According to SCE deterioration from erosion, over a period of 15 months, increased level of particles from disintegrated neutron absorbing panels in pool water by 134 %. These particles place an additional strain on pool water cleaning systems.

Equipment installed to make high-density pools safe exacerbates the danger of spent fuel cladding ignition, particularly with high burnup spent fuel. In high density pools at pressurized water reactors fuel assemblies are packed about nine to 10.5 inches apart, just slightly wider than the spacing inside a reactor. To compensate for increased risks of a large scale accident, such as a runaway nuclear chain reaction, pools have been retrofitted with enhance water chemistry controls and neutron absorbing panels between assemblies.

The extra equipment restricts water and air circulation, making pools more vulnerable to systemic failures. The ability remove decay heat from spent fuel pools to prevent boiling corresponds to the amount of water displaced in pool by spent fuel and equipment that allows for its tight packing. High density storage also impacts ability of water to flow through pool. If equipment collapses or fails, as might occur during a destructive earthquake or terrorist attack, air and water flow to exposed fuel assemblies would be obstructed, causing a fire, according to the NRC report Heat would turn the remaining water into steam, which would interact with zirconium, making problem worse by yielding inflammable and explosive hydrogen.

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made to the guidance in response to the comment.

However, as the comment is related to neutron absorbers in spent fuel pools, additional information on this topic may be found at:
(1) http://www.nrc.gov/waste/spent-fuel-storage/pools.html#nba, and (2) the draft Generic Letter on the subject of “Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools” located at http://pbadupws.nrc.gov/docs/ML1310/ML13100A086.pdf.

2.7 Comments from Richard Morgal

Comment 2.7.1

Comment: The lack of a cause and lack of immediate biological effect will ensure that a DSC breach at SONGS will never be attributed to any illness experienced by a driver on Interstate 5 experiencing an airborne radioactive release from a breached DSC. Just ask most cancer patients what was the cause of their disease; typically there is no known cause, it just occurred.

The pressure inside the DSC is only a few PSI above atmospheric pressure so there is a relatively small volume of gas with a high concentration of radioactive material that will be

released. Similar to a soft-drink being opened in an enclosed space, the initial release will quickly dissipate to an un-noticeable level in a matter of hours. Except that the CO2 is not deadly.

Response: This comment is not specific to NUREG-1927, and therefore, no changes were made to the guidance in response to the comment. See response to Comment 8.4.6.

Comment 2.7.2

Comment: I understand that DSC's that contain undamaged fuel rod assemblies will not contain any gaseous radiation, thus no radiation will be released when it breaches. The calculations and assessments requested throughout my comments should be made for DSC's that are known to contain damaged fuel rod assemblies, for which SONGS is known to have an exceptionally large number of damaged fuel rod assemblies. Maybe Dr. Kris Singh should be referenced as to where his predictions of millions of curie of radiation would be released from a microscopic breach of a stainless steel DSC.

Response: The staff recognizes that damaged fuel may be contained in a dry storage system, which requires consideration during the review of the renewal application. The design bases (as described in the Final Safety Analysis Report) for each dry storage system and ISFSI defines the required safety analyses for storage of damaged fuel. Damaged fuel is canned consistent with guidance in Interim Staff Guidance (ISG)-1, Revision 2 (see next paragraph). These safety analyses consider the configuration of the fuel can inside the cask/canister providing confinement. Therefore, the safety analyses are not dependent on the configuration of the fuel itself inside the can. Since reconfiguration of the can for damaged fuel may change the conclusions from safety analyses, it is therefore within the scope of renewal and subject to an aging management review.

ISG-1, Revision 2 provides guidance to the staff on classifying spent nuclear fuel as either (1) damaged, (2) undamaged, or (3) intact, before interim storage or transportation. From the guidance in ISG-1, Revision 2, spent fuel that has been classified as damaged for storage must be placed in a can designed for damaged fuel, or in an acceptable alternative. The purpose of a can designed for damaged fuel is (1) to confine gross fuel particles, debris, or damaged assemblies to a known volume within the cask, (2) to demonstrate that compliance with the criticality, shielding, thermal, and structural requirements are met, and (3) to permit normal handling and retrieval from the cask. The can also ensures compliance with 10 CFR 72.122(h)(1), which states that the spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. The can may also need to contain neutron-absorbing materials, if results of the criticality safety analysis depend on the neutron absorber to meet the requirements of 10 CFR 72.124(a).

As the guidance already addresses the scoping evaluation and the aging management review, as it would apply to either the fuel assemblies or a can for damaged fuel, the staff determined that changes to the guidance are not necessary as a result of this comment.

Comment 2.7.3

Comment: SCE is very interested in dismantling the current spent fuel pools ASAP and will be allowed to do so unless there are NRC documents that state spent fuel pools are an integral

portion of the AMP to address breached DSCs. The NRC should not allow any more nuclear power plants to be decommissioned until it is known what infrastructure residing on a recently shuttered nuclear power plant would be needed to repair or replace breached DSCs on-site.

This is a public safety issue, in that once one DSC breaches it is likely others will breach as well (same logic as is used in the lead canister approach that the NRC chooses to apply to the field, only the 1st breached canister is the lead canister). Once one DSC breaches, it would seem likely all other DSCs at the ISFSI would need to be evaluated to ensure further radiation releases were not eminent. All damaged DSCs would need to be repaired ASAP to reduce the likelihood of additional radiation releases. But if a spent fuel pool is required to repair faulty DSCs and there is no pool on site it could take years and 100's of millions of dollars to rebuild a pool that was hastily dismantled. Although the cost is an issue, the years it would take to replace the spent fuel pool while the public awaits the next canister to breach will not fair well on the nuclear industry's image.

Response: Comments on reactor decommissioning are outside the scope of NUREG-1927. The staff provides the following information to respond to the comment, but determined that no changes to the guidance were needed.

The staff notes that the use of a spent fuel pool is just one approach a licensee may take to address a canister breach. Some additional options may include: (1) conducting in-situ canister repairs, (2) creating a secondary confinement boundary, or nesting the breached canister within a new, larger confinement vessel, or (3) replacing the breached canister with a new canister, but engineering a shielded confinement structure to safely repackage the fuel rather than using a pool.

NRC staff is considering the issue of maintaining fuel handling capability at ISFSI sites as part of a separate effort and ongoing review of the storage and transportation regulatory framework, identified in COMSECY-10-0007 (http://www.nrc.gov/reading-rm/doc-collections/commission/comm-secy/2010/2010-0007comscy.pdf).

The comment makes an assertion, "Once one DSC breaches, it would seem likely all other DSCs at the ISFSI would need to be evaluated to ensure further radiation releases were not eminent." The goal of the aging management program is to identify and address any degradation *before* a loss of an intended function. The AMPs and associated aging management activities must allow for adequate time to identify aging effects before a loss of intended function. Also, the AMPs have elements related to corrective actions, confirmation process, and administrative controls (discussed in Section 3.6 of the guidance) that rely on the licensee's approved quality assurance program. The quality assurance program includes provisions to address the extent of condition (e.g., of any aging effects found) and preclude repetition of significant conditions adverse to quality.

2.8 Comments from Ray Lutz/Citizens' Oversight Projects

Comment 2.8.1

Comment: 1. TOO MANY DISTINCT AMPS: This document talks about how to design an Aging Management Plan (AMP) with the implication that potentially each holder of a specific license will have a separate and distinct AMP perhaps with almost no correlation to other sites. This makes it very difficult for the public and regulatory agencies to provide essential oversight

of the requirements in the AMPs because each one is different and requires individualized oversight and review. The variation between ISFSIs is not that great. Instead of a document that says how to design an AMP, the NRC should generate a standard AMP that is maintained as a public document.

Response: The NRC staff agrees in part with the comment. The NRC staff is currently developing a NUREG report that will provide an acceptable generic approach to the identification of credible aging effects in dry storage systems, and appropriate aging management activities needed to address these aging effects. This guidance is referred to as "Managing Aging Processes in Storage" (MAPS). MAPS will provide: (1) descriptions of storage systems, (2) technical bases for determining credible aging effects, (3) system-specific tables of subcomponents, their environments, and aging effects, and (4) generic aging management programs. Development of this document is expected to increase the efficiency of NRC staff reviews, by allowing the staff to focus its review on areas where applicants propose an alternative approach to the generic AMPs provided in the MAPS report. It is expected that this report will be published as a draft for public comment in summer of 2016, and the commenter may wish to review the MAPS report when it is issued for public comment.

However, the staff does not agree with an assertion made in the comment that it is difficult for regulatory agencies to provide essential oversight of the requirements in AMPs. The NRC currently provides effective oversight for all NRC-licensed activities, including individual ISFSI licenses and CoCs, and this will continue in the future.

No changes were made to NUREG-1927.

Comment 2.8.2

Comment: Licensees should have to publish their inspection data in a standard format so it is accessible and understood by the public so the public can provide oversight.

Response: The NRC staff disagrees with the comment. Licensees are required to record the results of their inspections and examinations and follow recordkeeping requirements for retention of those records, but the NRC does not require it to be recorded in a certain format. Also, such data is not typically submitted to the NRC. Rather, the records are subject to, and made available for, NRC inspection, as required in 10 CFR 72.82, "Inspections and Tests," and 10 CFR 72.232, "Inspection and Tests." The NRC has inspection manual chapters and inspection procedures for conduct of the inspections, and the NRC records the results of its inspections in inspection reports. The inspection reports are made publicly available, unless they contain classified, safeguards, or sensitive information. The commenter may find additional information on the NRC's inspection/oversight program of spent fuel storage at: http://www.nrc.gov/waste/spent-fuel-storage/oversight.html. No changes were made to the guidance.

Comment 2.8.3

Comment: The "hands off" approach being proposed here may have been appropriate for complex nuclear plants where each one is very different from other plants. Dry Storage systems are not so complex that a standardized AMP is not feasible.

Furthermore, we note that even for nuclear plants, Technical Specifications did move toward uniform standards rather than distinct specifications for each plant. There is no reason not to

establish uniform AMPs that can be shared as open standards rather than potentially having hundreds of distinct and incomparable programs.

Response: See the response to Comment 2.8.1.

Comment 2.8.4

Comment: 2. NO INFORMATION SHARING: The flow charts provided in this document describes how a licensee will design their own AMP and modify it based on information gained in their own review of their own program. It does not show how information is shared between licensees. Instead, as mentioned, the NRC should prepare a standard AMP which can be applied at nearly all dry storage sites. Any variation from the standard should be disclosed and approved. This will allow the information gathered from the sites to be compared so all sites can learn from lessons at other plants.

Response: The staff agrees with the comment. Information sharing between licensees is an essential tenet of the regulatory framework in terms of reporting, sharing, aggregating, and assessing operating experience across the industry, and outside of the nuclear industry, as it may be applicable. It is also an essential aspect of an effective "learning AMP" that considers and responds to operating experience and remains effective at managing aging effects in the period of extended operation. The NRC will inspect licensees' implementation of AMPs in the period of extended operation, including any licensee actions taken as part of the "learning" aspect of AMPs. The NRC will inspect licensees' periodic assessments of AMP effectiveness and any adjustments licensees have made to AMPs to respond to operating experience and ensure AMPs are effective for addressing aging effects in the period of extended operation. The use of an operating experience clearinghouse for sharing of aging-related dry storage system operating experience and other related information, is discussed in Section 3.6.1.10. The staff determined that additional changes are not needed to the guidance.

Comment 2.8.5

Comment: CONCLUSION: I sincerely hope this is not the direction the NRC is planning to take on this. Please create a standard AMP that will be a public document which is applicable to all spent fuel storage sites rather than this plan to establish no real standards, and end up with a bazillion difference AMPs which are nearly impossible to oversee.

Response: See the response to Comment 2.8.1.

2.9 Comment from Anonymous

Comment: Good

Response: The NRC staff notes the comment.

CHAPTER 3: RESPONSES TO PUBLIC COMMENTS ON GENERAL INFORMATION REVIEW

The comments on General Information Review relate to Chapter 1 of NUREG-1927, Revision 1.

3.1 Comments from Robert Einziger/NWTRB

Comment 3.1.1

Comment: P-11, L-8 - Clarify the statement starting with "the reviewer". If not in the SAR, state where the complete AMP will be preserved? The AMP is the cornerstone of this whole approach to the renewal assuring safety. Indicate the legal implications of not having the AMP in the CoC with respect to enforcement of the provisions.

Response: The complete AMPs are preserved in the renewal application, as supplemented. As discussed in Section 1.4.7, it is expected that license or CoC conditions will be issued as part of the renewal, requiring: (1) the licensee or CoC holder to incorporate a renewal supplement into the FSAR that includes the AMPs approved as part of the renewal, and (2) the specific or general licensee to develop procedures for AMP implementation. Having the AMP incorporated into the ISFSI or CoC FSAR and a licensee's procedures, as opposed to the license or CoC, allows the licensees the flexibility for adjusting aging management activities in the period of extended operation, to respond to operating experience and to ensure the effectiveness of the AMP. This flexibility is necessary for a "learning" AMP, as is discussed in Section 3.6.1.10 of the guidance.

However, the NRC staff also recognizes that there may be some aspects of AMPs that are critical to the NRC staff's safety findings of reasonable assurance of adequate protection of public health and safety, and the environment, and the decision to issue the license or CoC renewal. Such aspects may be included as terms, conditions, or specifications in the renewed license or CoC that will ensure the safe operation of the cask during the renewal term, as is discussed in Section 1.4.7 of the guidance.

The decision regarding what is included in the license or CoC versus in the FSAR or licensee's procedures will generally be made on a case-by-case basis. NRC staff will include essential safety-related aspects of the AMPs (upon which the NRC's regulatory decision was based) in the license or CoC, but that will not necessarily include the entirety of the AMP elements. This will allow for a learning AMP, where the licensees and CoC holders have the flexibility to modify their AMPs over time to respond to operating experience and ensure they are effective for addressing aging effects in the period of extended operation, using their programmatic review processes.

NRC staff also notes that if a safety issue is identified in the future, the NRC will require affected licensees to address the issue, even if it was not related to a particular condition in the license or CoC. Also, it is important to note that the licensee must meet all applicable 10 CFR Part 72 requirements, and the ISFSI or DSS SSCs must continue to meet their intended functions, in the period of extended operation.

The staff made several clarifications in Section 1.4.7 to reflect the discussion in this comment response.

Comment 3.1.2

Comment: P-11, L-27 -This is a repeat of line 1&2 on the same page.

Response: The information is applicable to both Sections 1.4.6 and 1.4.7, thus the repeat of the information is intentional. NRC staff concluded that no changes to the guidance were needed as a result of this comment.

3.2 Comments from Nuclear Energy Institute

Comment 3.2.1

Comment: Comment (Technical Specification/CoC content): The guidance document, and in particular Section 1.4.7, discusses the potential for inclusion of AMPs in the Technical Specification (TS) or as conditions in the renewed license or CoC. One of the key aspects of effective aging management and the "tollgate" approach contained in NEI 14-03 is the ability of any AMP to be updated as inspection results, research findings and operational data become available. Inclusion of an excessive amount of detail in the TS or CoC will restrict the ability of licensees and CoC holders to make, in a timely and effective matter, the updates which will be a vital part of a learning aging management program. Therefore, it is recommended that clear criteria be defined for the purposes of delineating the detail to be included in the TS, specific license or CoC versus the ISFSI storage system FSAR and the implementing procedures.

In 2013, NEI submitted a proposed rulemaking (PRM 72-7 - Petition for Rulemaking Submitted by NEI to Amend 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" (ML12299A380)) for the express purposes of defining criteria to determine the level of detail to be included in the TS of a specific system license or CoC. The defined criteria contained in the PRM lays a foundation for making a risk-informed assessment to determine the information to be included in the TS and license/CoC. The current guidance, which leaves the details of the information over which the NRC retains change control up to individual applicants and NRC reviewers—dependent on their ability to negotiate content on a case-by-case basis—does not provide for a stable, efficient or reliable regulatory environment.

10 CFR Part 54, "Requirements for Renewal of Operating licenses for Nuclear Power Plants," provides a worthwhile example of the appropriate level of detail to be contained in the license and TS with regard to license renewal and AMPs. For operating reactors, the TS contains a commitment to have an AMP (similar to a Quality Assurance Program, or an Emergency Preparedness Program) while the details of the program are contained in the FSAR and the licensee's implementing procedures subject to 10 CFR 50.59 change control. There are typically no changes to an operating plant's TS related to license renewal. We recommend that NUREG-1927, Revision 1, be constructed consistent with this precedent.

Response: The staff agrees in part with the comment. NUREG-1927 already draws from the 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," renewal process in terms of the appropriate level of detail to include in an NRC-controlled document versus a licensee-controlled document. Section 1.4.7 states that a specific license or CoC may include a condition to incorporate a renewal supplement into the FSAR.

However, there are two important differences in the reactor renewal framework and the storage renewal framework, which may lead to certain terms, conditions, and specifications to be included in individual 10 CFR Part 72 licenses and CoCs: (1) 10 CFR Part 72 does not include specific criteria for what type of information is included in technical specifications, as is done in 10 CFR 50.36, "Technical Specifications," and (2) 10 CFR Part 72 does not approve for incorporation by reference, consensus codes and standards, as is done in 10 CFR 50.55a, "Codes and Standards."

As discussed in the response to Comment 3.1.1, what is included in the license or CoC versus in the FSAR or licensee's procedures is determined on a case by case basis. NRC staff will include essential safety-related aspects of the AMPs (upon which the NRC's regulatory decision was based) in the license or CoC, but that will not necessarily include the entirety of the AMP elements. This will allow for a learning AMP, where the licensees and CoC holders have the flexibility to modify their AMPs over time to respond to operating experience and ensure they are effective for addressing aging effects in the period of extended operation, using their programmatic review processes. The staff made several clarifications in Section 1.4.7 to reflect the discussion in this comment response and the response to Comment 3.1.1.

The staff is undertaking several efforts that may allow standardization in the level and type of detail that is included in 10 CFR Part 72 licenses and CoCs. As noted in the comment, the staff has accepted PRM 72-7 for future rulemaking consideration. Also, the staff is currently developing a draft NUREG report (Managing Aging Processes in Storage (MAPS)) that will include an acceptable generic approach to the identification of credible aging effects in dry storage systems, and appropriate aging management activities needed to address these aging effects. The staff is also participating in the American Society of Mechanical Engineers (ASME) Section XI Task Group on In-Service Inspection of Spent Fuel Storage and Transportation Containments to develop consensus code case for in-service inspection of dry storage canisters. As these efforts are completed, the staff expects that there will be more standardization in the level and type of detail that is included in 10 CFR Part 72 licenses and CoCs. Also, NRC staff plans to update the guidance in NUREG-1927, as needed, so any future developments in these areas would be considered in a future update to NUREG-1927.

Comment 3.2.2

Comment: 5, 8-9. Add "(specific licenses only)" after the entries.

Response: The NRC staff agrees with the comment and made the recommended change.

Comment 3.2.3

Comment: 6, 14-15. Says 72.22 does not apply to CoC holders but "the reviewer should ensure" CoC holders submit the info in 72.22. 10 CFR 72.13 clearly stipulates information required for specific licensees, general licensees and CoC holders. To change the applicability of 72.22 to CoC holders should be implemented through rulemaking, not an NRC guidance document. It should be noted that this information is generally already contained in the certificate.

Response: The NRC staff agrees with the comment. The staff made clarifying changes throughout Sections 1.4.1, 1.4.2, and 1.4.3 to indicate where the guidance was applicable to specific license renewal applications only.

Comment 3.2.4

Comment: 8, 6. With respect to exemptions, does a CoC holder need to identify exemptions that General Licensees have taken to the CoC? Is there an opportunity to include wording in the renewed application that addresses exemptions and allows users to clear them from the licensing bases? Does this apply only to specific licensees?

Response: A CoC holder would need to identify any exemptions granted to it under 10 CFR 72.7, "Specific Exemptions." A CoC holder does not need to identify in its application, exemptions that general licensees have taken to the CoC. However, general licensees, in implementing any of the terms, conditions, and specifications of the renewed CoC (including implementation of the AMPs), do need to consider their specific licensing basis and the impacts of any exemptions on implementation of the AMPs and the renewed CoC. General licensees are expected to consider this and document it in their revised 10 CFR 72.212(b)(5) evaluation, as is discussed in Appendix E. In response to the comment, the staff added additional language to Appendix E for general licensees to consider their specific licensing basis and how any exemptions may impact aging management.

Comment 3.2.5

Comment: "9, 8-9. F-1, 14-15.

These sentences are not consistent with Appendix F, Figure F-1. Casks placed into service under an amendment not being renewed (either at the time of the first CoC renewal or subsequent CoC renewals) are allowed to stay in service for the full CoC term in effect at the time the cask is placed into service. Therefore, these casks need not be taken out of service when the CoC amendment under which they were loaded (initial or previously renewed) expires. Because this guidance addresses the first and subsequent renewals, we recommend deleting "upon expiration" from this sentence and adding to the end of the sentence "no later than a term of service equal to the initial or renewed CoC term in effect at the time the cask was placed into service."

Response: The NRC staff agrees with the comment. The sentence in question in both Section 1.4.4 and Appendix F were changed to read, "If [amendments/a CoC] are/is not renewed, upon expiration, casks loaded under that [amendment/CoC] would need to be removed from service when they reach the end of their storage term."

Comment 3.2.6

Comment: 9, 9. For casks loaded under an amendment not renewed, add an option to convert those casks to an amendment that is renewed per 72.212, prior to reaching the end of the storage term.

Response: The NRC staff agrees with the comment and has added the concept of applying changes authorized by another amendment, to Section 1.4.4.

Comment 3.2.7

Comment: 9, 21-25. Clarify whether the NRC expects applicants to supply non-proprietary versions of proprietary documents for docketing and public availability.

Response: The NRC staff clarified that applicants are expected to submit non-proprietary versions of the application documentation, in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Comment 3.2.8

Comment: 9, 27-28. Would licensing basis wording changes that do not affect the design bases be allowed by this statement? What changes are allowed? Relicensing is a major licensing activity and it represents an opportunity to clean-up certain kinds of issues that are not associated with the design bases but are associated with the license bases and creates problems for users.

Response: The language in question is meant to indicate that no changes to the design bases (which would typically be included in an amendment) should be submitted as part of the renewal application. The renewal application focuses on the current design bases and what aging management activities are needed in the period of extended operation to maintain the current design bases. Therefore, changes to the design bases should not be included as part of the renewal application, but should be addressed in a separate licensing process (like an amendment request). That being said, editorial clarifications or corrections are regularly included in applications, and the NUREG-1927 guidance is not meant to discourage submittal of those corrections or clarifications. The staff made a change to indicate that the renewal application may include editorial changes or corrections that do not change the design bases.

Comment 3.2.9

Comment: 11, 1-2. A license condition is not needed for future LARs to include aging management programs.

Response: The NRC staff disagrees with the comment. Once the ISFSI or CoC is renewed, this is the new licensing/design bases for the renewed ISFSI or CoC. Any post-renewal amendments to the license or CoC would need to address any effects on the renewed design bases, including any aging management programs or activities in effect. The guidance does not specify that future amendment requests must include AMPs, as the comment indicates. Rather, the guidance explains how a future amendment request can appropriately consider the renewed design bases of the license or CoC, including any approved AMPs and whether they encompass the new materials or SSCs introduced in the amendment. The staff clarified the language in Section 1.4.7 to reflect the discussion in this comment.

Comment 3.2.10

Comment: 11, 5. Recommend removing CoC. Conditions should only be added to the licensing basis (e.g., cask's FSAR) and not to the CoC.

Response: The staff disagrees with the comment. The regulations in 10 CFR 72.240(e) recognize that the NRC may revise the CoC to include terms, conditions, and specifications that will ensure the safe operation of the cask during the renewal term, including but not limited to, terms, conditions, and specifications that will require the implementation of an AMP. Please refer to the responses to Comments 3.1.1 and 3.2.1 for discussion on NRC-controlled versus licensee-controlled documents.

Comment 3.2.11

Comment: 11, 9-11. What is the purpose of reiterating the AMP in an Appendix of the SER? Documenting an AMP in the SER instead of the FSAR is not an apple-to-apples set of options. The SER is not subject to 72.48 and therefore could be interpreted as making the aging management program static. This would be counter to the goals of a learning program. It is recommended that this sentence be removed. PRM 72-7 includes a provision to remove review of the SER by the general licensee as a regulatory requirement.

Response: The purpose of including the AMPs in the NRC's safety evaluation report (SER) is for complete documentation of the staff's safety review, including recording the ultimate AMPs that formed the basis for the staff's safety findings and decision to issue the renewal. In addition, NRC staff reviewers and inspectors often refer to applicable SERs when conducting licensing reviews and preparing for and conducting inspections, as this is the complete record of the staff's evaluation, findings, and decision for licensing actions. The staff added text to Section 1.4.7 to explain this. Also, the staff does not consider documentation of the AMPs in the SER as making the AMPs static, as the comment indicates. In addition, although the NRC will consider PRM 72-7 in its rulemaking process, the requirement for a general licensee to review the SER (as provided in 10 CFR 72.212(b)(6)) must be met unless and until the regulation is revised.

Comment 3.2.12

Comment: 11, 14-15. There is no reason to have a license condition to require procedures. The licensees' QA programs require them to have procedures for safety-significant activities. Additionally, this sentence implies that the CoC holder would create implementing procedures, when this is a responsibility of the licensee.

Response: The staff disagrees with the first part of this comment. In the 10 CFR Part 54 reactor renewal process (that the commenter previously cited in Comment 3.2.1), the NRC typically includes a license condition as part of the renewal, requiring implementation of any new or revised programs or activities (related to the implementation of the AMPs included in the renewal application) to be completed within a certain timeframe. NRC staff feels this is an appropriate approach to also take for storage renewals. The staff made edits to the language in this section to indicate the condition may be tied to a particular implementation time. The staff agrees with the final part of the comment. The staff changed the sentence to clarify that the condition may be added to the CoC for the general licensee (cask user) to update, revise, or create procedures to implement the AMPs.

Comment 3.2.13

Comment: 11, 17. The specific edition and addenda of codes and standards approved as part of the amendment are part of the design basis (and therefore not subject to review). Provide clarification on the application of newly issued or revised codes and standards.

Response: The NRC cannot make a recommendation to incorporate future editions of codes or standards, given the uncertainty of their content. However, this does not necessarily prevent the incorporation of new standards into existing AMPs as they become available. First, the NRC expects to reference updated codes and standards in the upcoming MAPS report and its future revisions. Second, a licensee may consider incorporating new standards through its "learning AMPs," which are intended to evaluate new operating experience and information

and adjust aging management activities, as appropriate. To clarify this point, the staff removed the reference to codes and standards in Section 1.4.7, and added it to the list of sources of operating experience in Section 3.6.1.10 (that are considered by the applicant in development of its AMPs, or considered by a licensee in its learning AMP).

Comment 3.2.14

Comment: 11, 22-23. Define a "critical element" and get OGC concurrence if this is going to be the basis to include license conditions or Tech Spec requirements

Response: The staff has revised the language in Section 1.4.7 to more clearly describe what it means by a "critical element" of the AMP. See the responses to Comments 3.1.1 and 3.2.1.

Comment 3.2.15

Comment: 11, 27-28. This sentence duplicates lines 1 and 2 on the same page. See comment on 11 (1-2)

Response: See the response to Comment 3.1.2.

Comment 3.2.16

Comment: 11, 30. The term "as-needed" is vague. Any conditions should have a clear safety basis (preferably specific criteria) that establishes why the NRC needs to retain change control.

Response: The NRC staff agrees with the comment. The staff has revised the language in Section 1.4.7 to more clearly describe when terms, conditions, or specifications may be included in the license or CoC. The staff also deleted the referenced sentence, as it was duplicative of the guidance in this section. See responses to Comments 3.1.1 and 3.2.1.

Comment 3.2.17

Comment: 11, 32-33. How do license conditions "strengthen the technical basis for the reviewer to reach reasonable assurance"? The technical basis should be based on the TLAA and AMP, if needed.

Response: The staff agrees with the comment. The staff has revised the language in Section 1.4.7 to more clearly describe when terms, conditions, or specifications may be included in the license or CoC. See responses to Comments 3.1.1 and 3.2.1.

CHAPTER 4: RESPONSES TO PUBLIC COMMENTS ON SCOPING EVALUATION

The comments on Scoping Evaluation relate to Chapter 2 of NUREG-1927, Revision 1.

4.1 Comment from Robert Einziger/NWTRB

Comment: P-18, L22 - Since the event used in the NRC dry storage PRA was a drop event, provide the rationale for indicating that lifting rigs are excluded as important to safety.

Response: This section of the guidance (Section 2.4.3) references SSCs that are not typically included in the scope of the renewal review (e.g., lifting rigs), as this equipment may fall under the 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," design bases and not the 10 CFR Part 72 design bases for the ISFSI. In addition, the 10 CFR Part 72 design bases may vary between a specific licensee and a general license (that may use its 10 CFR Part 50 programs). However, the applicant is responsible for identifying, and the NRC is responsible for reviewing, what SSCs are within the scope of renewal, based on the individual design basis of the ISFSI or DSS. Although the guidance already includes a caveat that these items can be excluded from the scope of renewal provided that they do not meet scoping category (2) in Section 2.4.2, the staff also included a reference to scoping category (1), given the potential variance in individual design bases for ISFSI licensees and CoCs as discussed in this comment response. In addition, the staff added clarifying language to Section 2.4.2.2 to recognize the potential variance in individual design bases.

4.2 Comments from Nuclear Energy Institute

<u>Comment 4.2.1</u>

Comment: Comment (Application of aging management to temporarily used equipment, transfer cask and cask transporter): NUREG-1927, Section 2.4.2.2 discusses whether transfer casks and transporter devices are included in the scope of license renewal. The industry agrees that aging effects should be considered in assuring the capability of these items to perform any safety function they may have. However, it should be recognized in NUREG-1927 that these types of equipment are not in continual service and are more appropriately classified as "tools" with an Important to Safety function only when they are in use. Therefore, unless a unique aging mechanism is identified in the renewal application, AMPs for these items should simply refer to the normal periodic or pre-use maintenance and inspection programs already in place for the components. These programs, either described in the ISFSI or storage system FSAR or invoked via commitments to codes and standards (e.g. ANSI N14.6), have already been accepted by the NRC as adequate to ensure that the equipment is able to perform its design basis functions before being used. This approach ensures that inspections of such equipment are timely and are focused on when the equipment is being used to perform a safety function rather than conducting performing unnecessary inspections for equipment that is not in use.

Response: The staff disagrees with the comment. The staff notes that the decision to include an SSC within the scope of renewal is governed by the whether the failure of an SSC could impact an important-to-safety function, not whether adequate maintenance activities are in place. The guidance does not preclude an applicant from crediting maintenance activities to

manage the effects of aging, provided that the application includes appropriate technical justification. No changes were made to the guidance as a result of this comment.

Comment 4.2.2

Comment: 13, 22. Should there be a dot at the intersection of the column for "72.236 Applicable Sections" and the row for "SSCs not within the Scope of Specific License Renewal"?

Response: The staff agrees with the comment and made the recommended change. Although SSCs not within the scope of renewal are not explicitly discussed in 10 CFR 72.236, "Specific Requirements for Spent Fuel Storage Cask Approval and Fabrication," the review of this is inferred by the evaluation of SSCs within scope. This change is consistent with how 10 CFR 72.24," Contents of Application: Technical Information," is marked in the same table.

Comment 4.2.3

Comment: 13, 16. 46, Footnote.

The NRC has removed the terms "current licensing basis (CLB)" and "licensing basis" from the document as described in the footnote on page 46 and replaced that term with "design basis" on page 13 and elsewhere. The footnote further states that "The NRC does not believe that it is appropriate for the CLB to be applied to cask CoC renewals, which are generic." The industry disagrees that the term licensing basis does not apply to CoC renewals. While CoCs are generic, they establish the licensing bases for all general licensees using that cask design under the CoC. We also believe the term "licensing basis" is important for license and CoC renewals to establish the regulatory baseline for the application.

The industry does agree that the term "current licensing basis" may not be accurate for CoC renewals because all approved CoC amendments listed in 10 CFR 72.214 remain active and general licensees may use any approved CoC amendment. Thus, the industry believes both "licensing basis" and "design basis" are important terms to include in NUREG-1927 and NEI 14-03. The industry intends to add definitions of "Design Bases" and "Licensing Basis" to the Glossary in NEI 14-03. The definition of design bases will be taken from 10 CFR 72.3. The following definition of licensing basis, to be added the NEI 14-03 glossary, has been adapted from the NRC's definition of "current licensing basis" in 10 CFR 54.3:

Licensing Basis –The set of NRC requirements applicable to a specific ISFSI or DSS design and a licensee's or CoC holder's written commitments for ensuring compliance with and operation within applicable NRC requirements, and the ISFSI- or DSS-specific design basis (including all applicable modifications and additions to such commitments over the life of the license or CoC) that are docketed and in effect. The licensing basis includes applicable NRC regulations and appendices thereto; orders; license conditions; exemptions; and technical specifications. The licensing basis includes the ISFSI- and cask-specific design-basis information defined in 10 CFR 72.3 as documented in the latest ISFSI FSAR or applicable DSS FSAR revision for the ISFSI site. The licensing basis also includes the licensee's or CoC holder's commitments remaining in effect that were made in docketed licensing correspondence such as responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports.

Response: The staff concluded that no change was necessary to the guidance in response to this comment. The staff considers the proposed use of "design basis" in draft NUREG-1927,

Revision 1, as conveying the same information as that suggested by the commenter, even if not using the same terminology.

Comment 4.2.4

Comment: 14, 19-32. Add "approved exemptions" to the list for specific licensees.

Response: The NRC staff agrees and has made the recommended change.

Comment 4.2.5

Comment: 14, 33. NUREG/CR-6407 was issued in February 1996 which delineates the 3 levels of ITS, "A", "B" and "C". Older ISFSIs and CoCs have a basis that pre-dated this NUREG. A clarification statement should be added here about facilities that began operation prior to the publishing of NUREG/CR-6407, and more importantly that the basis of the safety classification is the information contained in the licensing basis documentation (i.e., FSAR, 72.48s, etc).

Response: NRC staff agrees with the comment, but concluded a change was not necessary. The staff noted that the bulleted list preceding the subject passage identifies the licensing documentation that can be used to verify the scoping evaluation for all ISFSI licenses and CoCs, regardless of age. Although NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," may be pre-dated by some licenses and CoCs, the staff still considers this document to be one resource for the understanding of how components are classified.

Comment 4.2.6

Comment: 18, 10. With respect NUREG/CR-6407, would a QA Category C item (minor impact on safety) be considered "important to safety"? Why not align this section with the three classifications in the NUREG to eliminate ambiguity for those applicants that use NUREG/CR-6407 as their basis?

Response: The staff notes that the determination of whether an SSC is within the scope of renewal is established by each licensee's design basis, and that design basis should not change in the renewal process. As such, there can be no generic determination of whether a QA Category C item is considered important to safety. No changes were made to the guidance as a result of this comment.

Comment 4.2.7

Comment: 18, 5. What is meant by fuel in a "disrupted state"? Where is this defined? Regardless of the state of the fuel assembly, the internals and fuel are part of the scope of the renewal.

Response: Section 2.4.2.1, "Scoping of Fuel Assemblies," has been revised to clarify how fuel assemblies may be considered in the scoping evaluation. Through the revisions to this section, the referenced statement has been removed.

Comment 4.2.8

Comment: 18, 4-5. Alternately, if a canister is intact (inert atmosphere maintained) and max cladding temperature & thermal cycling has been managed (both demonstrated for the initial license period in the FSAR and presumably revalidated during the relicensing application), wouldn't the spent fuel be in a known configuration (i.e., the same configuration it was in when loaded)? This was found to be an acceptable approach in previous license renewal applications.

Response: The comment is already addressed in Section 3.4.1.4 on the aging management review for fuel assemblies. Section 3.4.1.4 and the "Discussion" section in Appendix D provide clarification of the purpose of the example High Burnup Fuel Monitoring and Assessment Program presented in Appendix B. The staff notes that all issued renewals to-date include an AMP consistent with the guidance in Appendix D (i.e., Interim Staff Guidance, ISG-24). All specific licenses renewed prior to the issuance of ISG-24 did not include high-burnup fuel, and the staff used lessons learned from NUREG/CR 6745, "Dry Cask Storage Characterization Project—Phase 1; CASTOR V/21 Cask Opening and Examination," (Bare, et al., 2001), and NUREG/CR 6831, "Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage," (Einziger, et al., 2003) as confirmation that aging effects in low-burnup fuel do not result in changes to the analyzed fuel configuration, as defined in the approved design-bases of those ISFSIs.

The staff determined that changes to the guidance are not necessary.

Comment 4.2.9

Comment: 18, 27-30. SSCs associated with physical protection of the ISFSI are specifically excluded from review of the renewal request as illustrated in Section 2.2, yet in section 2.4.3, the reviewer is directed to verify that SSCs associated with physical protection of the ISFSI do not meet the scoping category in section 2.4.2. This should not be necessary since these SSCs are specifically excluded from review of the renewal request.

The same comment is applicable to SSCs associated with the ISFSI Emergency Plan.

Recommend deleting bullet 3 and 4 on page 18.

Response: The NRC staff agrees and has made the recommended changes.

4.3 Comment from Patricia Borchmann

Comment: In the Final NUREG 1927 (Rev. 2), the Scoping Evaluation will also need a corresponding expansion, to clarify how the expanded guidance applies to list of specified structures, systems, and components (SSCs), and will also apply to examine Concrete Containment structures which contain the dry cask storage cask containers inside the containment barriers. As another now known source of potential accelerated Concrete Degradation, the SSCs will require examination and analysis for presence of potential alkali-silica reaction (ACR), and define preventive measures, and define corrective actions.

Response: NUREG-1927, Revision1, Appendix B provides an example Aging Management Program (AMP) for Reinforced Concrete Structures, as these structures are generally within the scope of license renewal (following the scoping criteria defined in Chapter 2 of NUREG-1927,

Revision 1). This AMP includes 10 elements, each of which provides details for addressing aging effects for reinforced concrete structures (including those used as overpack structures for canisters). The 10 elements of the AMP describe the scope of the program (which identify all operable degradation modes, including the potential for alkali silica reaction), preventive actions, parameters monitored, detection of aging effects, acceptance criteria, monitoring and trending, as well as corrective actions and other quality assurance requirements.

The staff recognizes that the concern of potential alkali-silica reaction (ASR) degradation is of importance. Although, to date, no operating experience on ASR has been reported in any concrete structure used in dry storage systems, the staff recognizes the potential for ASR during extended periods of operation for the reasons identified in Information Notice (IN) 2011-20 (ADAMS Accession No. ML112241029). Therefore, the example AMP for reinforced concrete structures included in Appendix B assumes ASR as a potential operable degradation mode and includes activities for ensuring that ASR, if it occurs, is timely detected and addressed to ensure no loss of intended function for the concrete structure.

The staff determined that changes to the guidance are not necessary. In addition, the staff clarifies that concrete structures generally do not perform a confinement safety function in dry storage systems.

CHAPTER 5: RESPONSES TO PUBLIC COMMENTS ON AGING MANAGEMENT REVIEW

The comments on Aging Management Review relate to Chapter 3 of NUREG-1927, Revision 1.

5.1 Comments from Robert Einziger/NWTRB

Comment 5.1.1

Comment: P-34, L-29 - Add another bullet -"Determine that the corrective action will not exacerbate the degradation or create another aging issue on either the component in question or another component important to safety". In the reactor they thought they solved the baffle flow degradation issue by changing the flow pattern only to find that they only shifted the problem to another part of the fuel assembly. The issue continued at a higher rate and they had to go back and have another fix.

Response: NRC staff agrees and has made the recommended change. The intent of the bulleted list is to summarize, in part, elements of Corrective Action Programs consistent with Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," or 10 CFR Part 72, Subpart G, "Quality Assurance." The staff recognizes that not all licensees' CAPs may specifically address each of the bullets, and that is the basis for qualifying the list with "as applicable."

Comment 5.1.2

Comment: P-36, L-22 - Based on your definition of "operating experience" starting on the previous page and continuing through line 7 of this page, operating experience does not have occur for the cask or site in question. A new AMP may have significant operating experience to call upon. Suggest deleting or modifying the sentence starting on line 22.

Response: NRC staff agrees and has made the recommended change to delete the subject passage. The staff finds that the discussion immediately preceding the passage adequately addresses the evaluation of operating experience for new AMPS.

Comment 5.1.3

Comment: P-39, 1-14 - Based on the discussion in the previous paragraph, add the following at the end of line 14 "or time indicated in the CoC"

Response: NRC staff agrees and has made the recommended change.

Comment 5.1.4

Comment: P-36, L-13 -The use of "repository" in this context for experience-documentation is a poor choice. Suggest changing to something like "recognized and managed data base" or something similar.

Response: The staff agrees and has modified the language in Section 3.6.1.10.

5.2 Comments from Nuclear Energy Institute

Comment 5.2.1

Comment: The results of any evaluation should be made available for NRC inspection, as is the current practice. NRC's guidance should allow flexibility for licensees and CoC holders to demonstrate that sufficient information exists to determine the appropriate AMPs from previously conducted inspections and other sources of information (e.g., research, operating experience, monitoring, etc.).

Response: The staff agrees with the comment and has revised the guidance for conducting lead system inspections to recommend that pre-application inspections are one (but not the only) means of justifying AMP activities. The NRC responses to public comments on lead system inspections are located in Chapter 11 of this document.

Comment 5.2.2

Comment: Comment (Aging of equipment prior to being placed in service): Important to Safety equipment that is delivered to an ISFSI site and stored for some time before use is governed by the licensee's quality assurance program, which ensures the component can perform its intended function(s) use when placed into service. Specifically, 10 CFR 72.166 and 10 CFR 50, Appendix B, Criterion XIII both require measures to be established to control the "handling, storage, shipping, cleaning, and preservation of materials and equipment … to prevent damage or deterioration."

In accordance with these requirements, prior to placing a component in service, the licensee needs to ensure that the component is capable of performing its intended function. Therefore, NEI recommends that the NRC continue with the currently-accepted practice of determining the beginning of the storage term based on when the storage system is loaded with spent fuel and other authorized contents, and placed into service at the ISFSI pad. Any other method would create an overly complicated scenario where the end of the storage period could be dependent upon any of a number of factors (time of fabrication, period of time an empty storage canister or cask is stored on site, etc.).

Additionally, using the date when the system is placed in service would provide consistency with the dates provided to the NRC by the licensee as part of the notification requirements in 72.212(b).

It should be noted that precedence exists from power reactor construction and licensing for beginning the license term based upon when the plant entered commercial operation. Many reactors that were licensed under the 10 CFR 50 process applied for and were granted approval by the NRC to recover the construction period and begin the licensed term of the reactor based on when the reactor began operation. Given that many of the materials of construction and aging mechanisms are similar between reactors and dry cask storage systems, a strong precedent exists for beginning the licensed term for an individual dry storage system, based on when that storage system was placed in service.

Response: The staff agrees with this comment, and it is consistent with the guidance for the beginning of the storage term in draft NUREG-1927, Revision 1, Appendix F. The staff deleted the discussion in Section 3.2. However, the staff believes there is value in keeping the idea of consideration of the duration of time between the fabrication of a system and its deployment in

the ISFSI, in terms of selecting SSCs for inspection. The staff added this consideration to Section 3.6.1.10.

Comment 5.2.3

Comment: 22, Figure 3-1. The SSCs that are not subject to aging mechanisms do not need to be included in the FSAR supplement. There should be a separate screening AMR summary document outside the FSAR that lists the SSCs that were evaluated for aging and screened out. Only the in-scope SSCs should be included in the FSAR supplement.

Response: The staff does not make a specific recommendation for the location of the screening summary, provided that it is readily available for staff review. Section 3.4.1.2 of the guidance already notes that the applicant's determination of SSCs requiring no further review should be documented in the FSAR supplement or other application materials. The staff has changed the figure to state "Document in FSAR supplement or in the application."

Comment 5.2.4

Comment: 25, 16-18. This list of information requires clarification. Some of it is the subject of current requirements (such as radiation field), some of it is not. Wind, for instance, would be hard to summarize for many users. Also, how far back should it go? What is the reviewer expected to do with this data. What are CoC holders expected to provide when they have potentially many users? Recommend changing to "pertinent" environmental data that has a bearing on an AMP that is being recommended for a SSC. The amount of information that is needed to be provided by a CoC holder should be clearly understood.

Response: NRC staff agrees and has made changes to emphasize that information pertinent to aging is of interest. However, the staff does not consider it appropriate to identify the specific amount of detail needed for each type of data, as this is may depend on the specific application.

Comment 5.2.5

Comment: 25, 18. Change to "range of operating and service conditions."

Response: NRC staff agrees and has made the recommended changes.

Comment 5.2.6

Comment: 26, 5-9. Is it reasonable to request a CoC applicant to provide the requested environmental data for potential sites that the storage system is implemented? Change "evaluated" to "considered" on page 26, line 6.

Response: Proposed AMPs are anticipated to be applied to all sites using the CoC, and thus the provided environmental data necessarily must be shown to bound those sites. Also, the staff considers the term 'evaluated' to be appropriate and notes that this does not necessarily require deposition data specific for each site (as suggested in Comment 5.2.7).

Comment 5.2.7

Comment: 26, 6. Change to "the potential for deposition." Licensees should have the option of addressing degradation mechanisms via aging management programs without having to gather specific environmental and deposition data prior to submitting an application.

Response: NRC staff agrees and has made the recommended changes.

Comment 5.2.8

Comment: 29, 21-22. An analysis to justify not including in aging management activities is a TLAA by any other name and this approach seems to violate 72.3 definition of TLAA (6). This wording is confusing as it is not clear whether "analyses that are not TLAAs" can be used to justify the frequency of or not needing an AMP.

Response: The staff makes clear that these analyses are not TLAAs by definition, but may be submitted in support of the proposed aging management programs. The staff revised the language to provide further clarification.

Comment 5.2.9

Comment: 32, 37. Please clarify what is meant by: "The method should be adequate and proven…" What does proven mean? Demonstrated? Qualified?

Response: NRC staff has revised the passage to clarify the intent of the guidance.

Comment 5.2.10

Comment: 33, 16-19. Suggest deleting "Data Collection." It does not add any value.

Response: The staff disagrees with this comment and did not make a change. The staff determined that the guidance for data collection was appropriate, considering that adequate documentation of inspection results is an integral part of the period reviews (e.g., "tollgates") of AMP effectiveness.

Comment 5.2.11

Comment: 34, 3. It is impossible to "avoid" using non-quantifiable terms. Suggest using "minimize."

Response: NRC staff concluded that no change was necessary because the intent is conveyed by the use of either term.

Comment 5.2.12

Comment: 34, 23. Revise to "… and includes provisions (as appropriate) to:" Not all of these activities are performed, depending on the issue being addressed in the corrective action process.

Response: NRC staff agrees and has made the recommended changes.

Comment 5.2.13

Comment: 34, 33-36. This paragraph requiring all applicable corrective actions to be discussed in the application appears to contradict lines 16-23, which (correctly) defers to the licensee corrective action program (CAP). Specific corrective actions will depend on the problem found, severity, and safety significance. The aging management program should not be used to supersede the licensee corrective action program.

Response: NRC staff agrees, in part, with the comment and removed the recommendation to discuss all corrective actions in the application. However, the staff finds that, in some cases, it may be appropriate for an AMP to contain specific corrective actions that supersede the CAP. The discussion on corrective actions was revised to reflect this.

Comment 5.2.14

Comment: 35, 12. Suggest changing "canister" to "carbon steel components." Coating a canister for aging management of corrosion is not a good example of confirming the acceptability of corrective actions.

Response: NRC staff agrees with the comment and has made the recommended change.

Comment 5.2.15

Comment: 35, 11-13. A canister in storage, horizontal resting on rails would be difficult to coat. Does the NRC mean the overpack vs. canister?

Response: See the response to Comment 5.2.14.

Comment 5.2.16

Comment: 35, 17-26. Should state that the following are examples of provisions (Lines 21 to 22) as other Appendix B criterion apply to the AMP, e.g. Test Control, etc. which are not listed.

Response: NRC staff agrees and has made the recommended changes.

Comment 5.2.17

Comment: 35, 17-29. This section implies programs are to be included in the FSAR supplement. Please clarify. Programs are not typically included in FSARs. Most licensees and CoC holders perform these actions as a part of their NRC-approved QA Program required procedure set. Suggest this section be simplified to verifying that the applicant has an NRC-approved QA Program.

Response: NRC staff agrees and removed the guidance for the staff to verify aspects of the QA program, as QA programs necessarily have been approved by the NRC. Also, the second sentence of this section was revised to clarify that aging management programs must be administratively controlled and included in the FSAR.

Comment 5.2.18

Comment: 35, 27-28. Delete (a). Reporting inspection results to the NRC would be no different than any other activity and NRC reporting is governed by existing regulation (10 CFR 72.75), not an AMP.

Response: NRC staff agrees and has made the recommended changes. The staff also found that part (b) of this paragraph is adequately addressed in Section 3.6.1.10., and thus that passage was removed as well.

Comment 5.2.19

Comment: 36, 2, 28. This "relevant international and non-nuclear operating experience" item is part of the minimum expected constituents of the OE to be evaluated. It is not reasonable to expect, and likely not even possible to achieve, absolute knowledge of all relevant international and non-nuclear OE. This should be changed to "any known relevant international and non-nuclear operating experience." This issue will be addressed in the operating experience sharing program proposed in NEI 14-03.

Response: NRC staff disagrees with the comment and did not make a change. It is the NRC staff's expectation that the staff will maintain a general, current knowledge of international and non-nuclear operating experience, and consider this information when reviewing the application.

Comment 5.2.20

Comment: 36, 8-13. This paragraph can be deleted as it is covered in the operating experience sharing section of NEI 14-03. The methods to obtain, aggregate, and enter specific and industry-wide operating experience will be defined.

Response: Although NEI 14-03 may ultimately provide one acceptable means to meet the intent of this activity, the staff has not yet endorsed NEI 14-03, and regardless, other approaches may be acceptable. The staff did update this discussion to include a reference to the ISFSI Aging Management Institute of Nuclear Power Operations Database, which is discussed in NEI 14-03, Revision 1, as a potential framework for operating experience that the applicant may reference.

Comment 5.2.21

Comment: 36, 22. Change "an licensee's" to "a licensee's"

Response: This comment was superseded by the resolution of Comment 5.1.2, which deleted this passage.

Comment 5.2.22

Comment: 36, 27. Under the section for Learning AMPs, it states that the NRC reviewer "should ensure that applicants commit to future reviews of OE." This seems to be a redundant requirement, since AMPs will need to include a section for OE that will cover experience capturing and assessment. It's not clear how applicants (particularly CoC holders) would commit to future reviews of OE beyond current established procedures (both CAP and OE program). This concept will be captured in the tollgate assessments contained in NEI 14-03

Response: Because of this and other comments, the Operating Experience discussion in Section 3.6.1.10 was revised to clarify the guidance and reduce redundancies.

Comment 5.2.23

Comment: 36, 33. Change "deficiencies" to "lessons learned."

Response: NRC staff agrees and has made the recommended changes. The staff also found that part (b) of this paragraph is adequately addressed in Section 3.6.1.10., and thus that passage was removed as well.

Comment 5.2.24

Comment: 37, 1-2. NEI 14-03 will be Revision 1 and will have a new title: "Format, Content, and Implementation Guidance for Dry Cask Storage Operations-Based Aging Management"

Response: Although the staff has not yet endorsed NEI 14-03, the staff made changes to Section 3.6.1.10 to include updated discussion on certain concepts contained in NEI 14-03, Revision 1.

Comment 5.2.25

Comment: 37, 15-16. Tollgates are not intended to be a part of the Tech Specs or license conditions.

Response: The staff disagrees with the comment and did not make a change. The staff may consider the inclusion of tollgates as a condition of the license or CoC, as necessary, to reach reasonable assurance that operating experience will be appropriately evaluated. This approach is consistent with that used in reactors, where the sharing of OE became a requirement that arose out of the TMI action plan.

Comment 5.2.26

Comment: 37, 17-22. Updated NEI 14-03 will include a more comprehensive description of an operating experience sharing program. Each applicant should not have to provide details about the industry operating experience sharing program.

Response: Although the industry clearinghouse described in NEI 14-03 may provide one acceptable means to meet the intent of this activity, it is expected that the application will still contain a general description of the framework for OE sharing, citing the INPO clearinghouse if used. See the response to Comment 5.2.20. The staff also notes that it has not yet endorsed NEI 14-03.

Comment 5.2.27

Comment: 38, Figure 3-2. Figure 3-2 should be deleted. The flow diagram is just a listing of the 10 AMP elements, with no apparent ties to their use in a flow chart. For example "corrective action" is a decision box with only one output. In the sense of a true learning AMP, external and internal OE is primarily input that could be used to inform any of the various elements of a given AMP.

Response: NRC staff agrees and has deleted the chart, as it was found not to add value to the discussion.

Comment 5.2.28

Comment: 39, Section 3.6.2, 3.6.3 and Appendix E. These section discuss the need to develop the AMP infrastructure prior to entering the period of extended operation, but does not go into details about the necessary framework or acceptable implementation schedule (other than the note that the infrastructure for AMP implementation should be no later than one year from the date the NRC issues a renewed Specific license or CoC). Having limited guidance and short time frame may be very challenging for those general licensees that have loaded early on in the initial CoC term, and will most likely be under a period of timely renewal for the application submittal, depending on the length of NRC review. To ensure that general licensees have sufficient time to budget, plan resources and conduct a relevant inspection it is recommended to change this requirement to eighteen months

Response: NRC staff revised the discussion on AMP implementation in Section 3.6.3 to provide more clarity with regard to timely renewal and longer implementation schedules. The staff did not change the guidance for developing the infrastructure for AMP implementation generally within 1 year from the date the NRC issues the renewal, as the guidance recognizes that longer schedules may be proposed and justified in the renewal application.

Comment 5.2.29

Comment: 39, 16-25. General licensees that loaded soon after a CoC was made effective will not be able to implement AMP infrastructure prior to extending the period of extended operation, because the renewal application and any proposed AMPs will be still under review by the NRC. One year for general licensees in timely renewal to implement AMPs is a short timeframe for budgeting and planning resources to ensure that reliable inspection capabilities are qualified and available. Recommend changing one year to two years.

Response: See the response to Comment 5.2.28.

5.3 Comments from Donna Gilmore/SanOnofreSafety.org

Comment 5.3.1

Comment: 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.

Response: The staff considers the recommendation of specific repair/remediation activities to be most effectively addressed on a case-by-case basis, given the difficultly of defining generic guidance to address a variety of potential conditions. The staff concluded that no change was necessary.

Comment 5.3.2

Comment: 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.”

Response: The staff disagrees with the comment and did not make a change. The staff recognizes that licensees will need time to develop procedures and coordinate inspections, and that process can only begin after the NRC approves an applicant’s approach to manage aging by issuing the renewal.

CHAPTER 6: RESPONSES TO PUBLIC COMMENTS ON GLOSSARY

The comments on the Glossary relate to Chapter 5 of NUREG-1927, Revision 1.

6.1 Comments from Robert Einziger/NWTRB

Comment 6.1.1

Comment: P-46.L-30 -There is no regulation or guidance that requires the use of an inert atmosphere to dry store SNF. It is recommended in NUREG-1536, but not required. CANDU fuel is stored in air. If the temperature is low enough, or the condition of the fuel or cladding doesn't matter such as the storage of fuel rod fragments, PWR fuel could also be stored in air or nitrogen.

Response: The NRC staff agrees with the comment in terms of there being no specific requirement to store spent fuel in an inert atmosphere. The staff modified the definition of "dry storage" and "dry storage system" accordingly.

Comment 6.1.2

Comment: P-48, L-4 - Define "moderate": its use goes counter to the guidance in this document not to use unquantifiable terms.

Response: NRC staff agrees with the comment. A quantification of "moderate" was provided in the definition of "off-normal events or conditions," consistent with the guidance in NUREG-1536.

Comment 6.1.3

Comment: P-48, L-7 - In the initial application the applicant is supposed to analyses off-normal events per 10 CRF 72.236(l) "The spent fuel storage cask and its systems important to safety must be evaluated, by appropriate tests or by other means acceptable to the NRC, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions". If the analysis showed permanent deformation but no implication for safety, under the wording in this sentence the off-normal event would have to be classified as an accident. Reword the sentence.

Response: NRC staff agrees with the comment. The definition of "off-normal events or conditions" was revised to remove the wording in question, as it is not necessary to define the term.

Comment 6.1.4

Comment: P-48, L25 -This is not a definition of "retrievability" just a statement of why it is necessary. Provide a definition of "retrievability"

Response: The staff disagrees with the comment. The definition cites the regulatory requirement for "retrievability" (10 CFR 72.122(l)) and the guidance (ISG-2) that describes retrievability and what is expected in terms of ready retrieval to meet the regulatory requirement.

Rather than repeat the guidance of ISG-2 in the definition, the staff believes it is appropriate to cite the guidance in the definition. No changes were made to the guidance.

6.2 Comments from Nuclear Energy Institute

Comment 6.2.1

Comment: 47, 29. Should "testing" be more specific? Does monitoring include all gathered data including AMP inspections, or only data from non-aging management activities such as TS surveillance, FSAR-required annual inspections, etc.?

Clarification is requested on what testing is being referred to? How does the referred-to testing differ from monitoring?

Response: Monitoring is meant to include all gathered data from aging management and "non-aging-management" activities (e.g., TS surveillances required in the initial license or CoC). The staff revised the language of the "Monitoring" definition in Chapter 5 to clarify.

Comment 6.2.2

Comment: 47, 34. "Controlling water chemistry" is not a good example for DSS because during storage, there is no water chemistry to control. (Wet ISFSIs are specifically excluded from this guidance.) Is this intended to mean groundwater chemistry? Suggest changing to "monitoring water chemistry."

Response: NRC staff agrees with the comment. The clause was moved and the language clarified.

Comment 6.2.3

Comment: 47, 40. "No temporary or permanent degradation…" is too restrictive. Some degradation, especially temporary, is expected and may occur. AMPs are being put in place to ensure degradation is managed and mitigated, if necessary. Recommend deleting "and to experience no temporary or permanent degradation from normal operations, events and conditions."

Response: NRC staff revised the definition of "normal events or conditions" to remove the wording in question, as it is not necessary to define the term (similar to the change made to the definition of "off-normal events or conditions," as discussed in the response to Comment 6.1.3).

Comment 6.2.4

Comment: 48, 17. Change "design" to "Certificate of Compliance." CoCs, not designs, are renewed.

Response: NRC staff disagrees with the comment, as the language in the "Renewal of a License or CoC" definition in the guidance is consistent with the language in 10 CFR 72.240, "Conditions for Spent Fuel Storage Cask Renewal." No changes to the guidance were made.

Comment 6.2.5

Comment: 49, 8. What is the purpose of referring to NUREG-1536 for just the definition of retrievability? Retrievability is contained in 10CFR 72

Response: The reference to NUREG-1536 was included to covey the source of the definition for "safety function." The reference to NUREG-1536 was repositioned, so that it does not appear to only apply to retrievability.

Comment 6.2.6

Comment: 49, 23. Change "features" to "SSCs."

Response: NRC staff disagrees with the comment, as the language in the "Structures, systems, and components important to safety" definition in the guidance is consistent with the language of this definition in 10 CFR 72.3, "Definitions." No changes to the guidance were made.

CHAPTER 7: RESPONSES TO PUBLIC COMMENTS ON NON-QUANTIFIABLE TERMS

The comments on Non-Quantifiable Terms relate to Appendix A to NUREG-1927, Revision 1.

7.1 Comment from Nuclear Energy Institute

Comment: A-2, The first bullet in the right column. This bullet states that quantitative information must be provided if a non-quantifiable term is used. This is confusing. If a quantifiable term can be used it should be used. Can the NRC clarify?

Response: Table A-1 is intended to provide guidance for the use of non-quantifiable terms in applications. The bullet in question is meant to convey what the commenter suggests. Non-quantifiable terms that "screen in" (as discussed in Table A-1) should either be substituted with a quantifiable term or quantitative information, or if it cannot be quantified, it should be further described or defined. No changes were made to Table A-1. However, a change was made to page A-1 to note that it is preferred to use quantifiable terms and quantitative information where it exists.

CHAPTER 8: RESPONSES TO PUBLIC COMMENTS ON EXAMPLE AMP FOR LOCALIZED CORROSION AND STRESS CORROSION CRACKING OF WELDED STAINLESS STEEL DRY STORAGE CANISTERS

The comments on Example AMP for Localized Corrosion and Stress Corrosion Cracking of Welded Stainless Steel Dry Storage Canisters relate to Appendix B, Table B-1, of NUREG-1927, Revision 1.

8.1 Comment from Ace Hoffman

Comment: The nuclear industry says not to worry about rust, not even in coastal dry cask storage zones, even though salty air is highly capable of causing Chloride Induced SCC. In fact, the nuclear industry's own experience has been that cracks can develop all the way through a 1/2 inch stainless steel sheet of metal in less than two decades, and cracks can start to form within a few years. Yet the nuclear industry plans to let these thin storage canisters sit in a highly corrosive environment for up to six decades without a blush, and probably a lot longer.

Inspecting for SCC is virtually impossible, especially around the load-bearing points, the welds, or anything on the inside of the canister.

Response: NRC regulations require that dry cask storage systems must be amenable to inspections. Nondestructive examination (NDE) methods that could be used to inspect canisters already exist and have been used by the nuclear industry for decades to examine structures, systems, and components that are constructed from welded stainless steels and welded austenitic nickel-based alloys. Methods to apply existing NDE techniques to welded austenitic stainless steel canisters have been developed and are being tested by both the Electrical Power Research Institute (EPRI) and dry storage system manufacturers. The NRC and the nuclear industry, including spent fuel storage system manufacturers and users of these systems, are active participants in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Task Group to develop a code case for the examination, acceptance criteria, and corrosion/degradation assessment methodologies for dry cask storage systems.

The NRC requires licensees and CoC holders to evaluate potential aging effects for dry storage systems, and where necessary, include aging management programs to assure that the important to safety structures, systems, and components continue to maintain their intended functions. If a specific inspection or examination methodology is under development at the time of a renewal review, the NRC may condition a license or CoC renewal to require development of the methodology within a certain timeframe. If the licensee or CoC holder identifies that proposed inspections cannot be performed in accordance with conditions of the license or CoC renewal, then the licensee or CoC holder is out of compliance with the renewed license and would be required to propose alternatives that demonstrate the safety functions of the structure, system, or component that is potentially subject to aging will be maintained for period of extended operation. NRC staff determined no changes were necessary to address this comment.

8.2 Comments from Nuclear Energy Institute

Comment 8.2.1

Comment: Comment (Reliance on the licensee's Corrective Action Process): The example AMP on CISCC in Appendix C includes an extensive discussion of the expected corrective actions, including specific, numerical increases in the sample size and inspection frequency, and use of more rigorous inspection methods. It is the role of the licensee to use its Corrective Action Program (CAP) to determine the appropriate corrective actions needed, including making changes to inspection frequencies, techniques, and sample sizes. The corrective actions stated in the example AMP are too detailed and should be simplified to ensure that licensees have sufficient flexibility to determine the appropriate corrective actions and extent of condition for their site. At a minimum, wherever numerical values are shown in the sample AMP (e.g., sample size increases), they should be shown in brackets, indicating that these values are examples and the licensee or CoC holder should determine the appropriate value. The corrective actions description on page B-16 and B-17 associated with reinforced concrete structures are reflective of the appropriate level of detail for corrective actions associated with an AMP.

Response: The NRC staff partially agrees with this comment. The AMP provided in Appendix B is an example. A licensee or CoC holder can use the example or create its own AMP and provide a supporting technical basis for the AMP. The CISCC example AMP provided in Appendix B relies on the inspection criteria in ASME Boiler and Pressure Vessel Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," including, to the extent practical, inspection methods and acceptance criteria. Corrective actions in the CISCC example AMP include examples of subsequent and supplemental inspections when acceptance criteria are not met. These subsequent and supplemental inspections are consistent with the requirements for inspections of nuclear power plant components in ASME B&PV code Section XI. NRC staff note that ASME Section XI has formed a task group to develop a code case for inservice inspections of dry storage systems. The code case will include inspection requirements and acceptance criteria. NRC staff is participating in the task group and expects that a code case to be developed within 5 years. Applicants for renewals, including specific licensees and CoC holders, may choose to use an NRC accepted ASME code case to support AMPs. With respect to the Reinforced Concrete AMP in Appendix B, the NRC staff note that the example AMP points to applicable concrete rehabilitation guides or standards including ACI 224.1R and ACI 364.1R, both of which are referenced by ACI 349.3R. Note that ACI 364.1R contains specific guidance on performing preliminary and detailed investigations for the assessment of a concrete structure's behavior or condition, including the information that should be reviewed and when a detailed investigation is warranted. The NRC staff made multiple changes to the corrective actions portion of the CISCC AMP to address this comment and additional NEI comments.

Comment 8.2.2

Comment: Comment (Reporting requirements): The example AMPs in Appendix B provide several instances where a licensee would provide a report to the NRC based upon the results of an inspection and the potential corrective actions. It is important for licensees to have controls and programs in place for reporting results to the NRC. 10 CFR 72.75 already provides reporting requirements for specific and general licensees, ranging from one-hour to eight-hour verbal reports, with follow-up written reports. If additional reporting requirements are to be included in a licensee's program, rulemaking should be undertaken to modify 72.75. It should

be noted that any inspection reports and periodic tollgate assessments can be reviewed as part of the NRC's inspection and enforcement process. NEI 14-03 will provide additional details on the industry operating experience sharing program.

Response: NRC staff agrees with the comment and has made changes to the example AMPs to remove reporting requirements. The licensees' aging management activities and results will be available for NRC inspection. Also, licensees will report aging-related operating experience to an operating experience database or clearinghouse.

Comment 8.2.3

Comment: B-2, Scope of Program. Recommend deleting the fifth bullet. Some horizontal surfaces (the top on a vertically oriented canister) are the least susceptible area on the can (hottest, thickest). They are probably the easiest surfaces to inspect, but not highly valuable or significant to the safety of the system.

Response: The NRC staff partially agrees with the comment. Areas where atmospheric deposits tend to collect are more likely to have accumulations of potentially aggressive species that may promote corrosion. Although the top surfaces are expected to be at higher temperatures than the lower surfaces, the actual temperature at the top of the cask is dependent on several factors including initial thermal loading and time in service. Thermal models have also shown that there can be a significant temperature gradient along the top of a horizontally positioned canister with the ends of the canister being at a lower temperature compared to the center of the canister. These factors should be considered in the selection of the system for inspection as described in AMP Element 4: Detection of Aging Effects/Sample Size. The NRC notes that some canister designs have through thickness circumferential welds in the shell of the canister near the end. For these canisters, the combination of (1) a horizontal surface that can collect deposits, (2) cooler temperatures that can promote deliquescence, and (3) the presence of a weld that is likely to have through thickness tensile stresses that can promote crack growth exists and should be considered as an area of interest for inspection. The NRC staff added a statement indicating areas where a combination of attributes (as discussed above) exist should be considered and identified as a higher priority location for inspection.

Comment 8.2.4

Comment: B-3, Detection of Aging Effects. The example AMP provides a very low threshold to performing volumetric examination (i.e. evidence of localized attack (pitting), even away from a weld). Therefore, any evidence of aging mechanisms would lead to volumetric or surface examination, which is not reasonable.

Response: The NRC staff partially agrees with the comment. Visual inspection systems exist that have provisions for measuring both the size (i.e., surface area) and depth of localized indications. Provided that there is sufficient access to and cleaning of areas where localized corrosion is suspected, visual inspection performed using such systems is sufficient to evaluate whether pitting corrosion has occurred and determine the depth of pitting attack.

The NRC also recognizes that the conditions necessary for pitting corrosion and stress corrosion cracking in chloride environments are similar, and that pitting corrosion has been shown to be a precursor to chloride induced stress corrosion cracking (CISCC). The probability of detecting cracks using visual methods depends on both inspection parameters and crack

opening displacement. Visual methods cannot be used to detect the depth of cracks. Because of the possibility that pits may be initiation points for CISCC and the difficulty detecting cracks using visual methods, surface or volumetric methods are necessary to determine if cracking is also present when pitting or crevice corrosion has been identified.

NRC staff clarified the guidance in the AMP to reflect the above discussion. Specifically, the staff revised the visual examination section to include the use of visual methods to measure the size and depth of pitting. Specific additions indicate that the measurement methods should be demonstrated using pits on control samples where the pit depths are independently measured using an independent method such as ultrasonic testing. The staff also included guidance on: (1) use of visual examination to identify if cracking has initiated, (2) volumetric examination is necessary if evidence of cracking is identified by visual examination, and (3) surface and volumetric examination is necessary if visual examination cannot be used to assess whether cracking has initiated from pitting. The NRC and the nuclear industry, including spent fuel storage system manufacturers and users of these systems, are active participants in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Task Group which is working to develop a code case for the examination, acceptance criteria, and corrosion/degradation assessment methodologies for dry cask storage systems. Once the code case is developed, the NRC will review it for potential endorsement and approval for use to meet regulatory requirements. If the staff approves the code case for use, it may include a reference to it in a future update to NUREG-1927 or in the Managing Aging Processes in Storage Report, which is in development.

Comment 8.2.5

Comment: B-4, Detection of Aging Effects. The requirement to implement surface and/or volumetric upon discovery of localized corrosion (e.g. pitting) is impracticable. While CISCC often initiates at the base of pits, most pits are not initiators of CISCC. Suggest these methods only be required when the observed localized corrosion exceeds the acceptance criteria of IWB-3640.

Response: The NRC staff generally disagrees with the comment. See the response to Comment 8.2.4.

The acceptance criteria in IWB-3640 requires characterization of the flaw including flaw dimensions that would require information that may necessitate volumetric examination. Given the relatively small area affected by pitting corrosion, it is unlikely that pitting damage would exceed the acceptance criteria using the evaluation method defined in IWB-3640 unless (or until) the pit depth exceeded 75 percent of the wall thickness. Initiation of cracks from pits can occur in pits that can be less than 100 microns deep. Thus, the need for surface and/or volumetric inspection is not driven by the size or depth of the pits, but rather the potential for crack initiation.

The NRC recognizes that the key difference in the conditions necessary for chloride-induced pitting and CISCC is the presence of a tensile stress necessary for crack initiation and propagation. Pitting may occur on any surface where chloride containing salts may accumulate and undergo deliquescence. In contrast, CISCC is expected to be limited to welds and weld heat affected zones where residual tensile stresses are present. Based on this understanding, the Scope of Program specifically states that examinations should be focused in those areas.

NRC staff changed the detection of aging effects element to specify volumetric examination for cracks of pits that are located within 25 mm (1 inch) of a weld. Pitting on surfaces that are known not to have either a fabrication, closure, or temporary attachment weld do not require examination for cracks. The NRC and the nuclear industry, including spent fuel storage system manufacturers and users of these systems, are active participants in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Task Group which is working to develop a code case for the examination, acceptance criteria, and corrosion/degradation assessment methodologies for dry cask storage systems. Once the code case is developed, the NRC will review it for potential endorsement and approval for use to meet regulatory requirements. If the staff approves the code case for use, it may include a reference to it in a future update to NUREG-1927 or in the Managing Aging Processes in Storage Report, which is currently in development.

Comment 8.2.6

Comment: B-4, 1st paragraph. Since efforts to define ASME Section XI requirements for dry storage are just beginning, flexibility needs to be maintained for future improvements. Suggest revising the end of the sentence to "…inspection techniques consistent with available and applicable codes and standards."

Response: The NRC staff agrees with the comment. Section 3.6 of NUREG-1927 identifies the guidance for reviewing AMPs. Specifically, section 3.6.1.4 identifies applicable consensus codes and standards for the detection of aging effects. The use of consensus codes and standards is also identified in section 3.6.1.6 Acceptance Criteria. The AMP provided in Appendix B is an example AMP. As an example, the AMP has identified methods for the detection of aging effects, specified acceptance criteria, and corrective actions required if the acceptance criteria are not met. As stated in the example, the AMP is based on examination methods identified in Section XI of the ASME code for pressure boundary components.

The NRC is actively involved in the ASME Task Group assembled for the purpose of developing a code case for the inservice inspection requirements and acceptance criteria for dry cask storage systems. Although the NRC supports the development of a code case, the time for development is expected to be 5 years. Once the code case is developed, the NRC will review it for potential endorsement and approval for use to meet regulatory requirements. Based on NRC staff experience in specific license and CoC renewals, a revision to the guidance in NUREG-1927 was required before an acceptable code case could be developed and approved by the NRC. In the future, the guidance in NUREG-1927 may be revised to reference an NRC approved code case rather than the currently referenced requirements developed for inservice inspection of nuclear power plant components.

In response to this comment, NRC staff added information on the AMP and the potential development of a code case for the inspection of dry cask storage systems to the introductory paragraphs preceding Table B-1. The staff also revised the introductory paragraph to include statements that the NRC accepts inspections based on NRC approved code cases or NRC-endorsed revisions of consensus codes and standards that are specifically applicable to spent fuel dry storage systems.

Comment 8.2.7

Comment: B-4, Sample Size. Need to allow for bounding sites in the future. Recommend changing to: "Initially one canister at each site until sufficient data and experience are gathered to justify bounding a site with inspection results from another site."

Response: The NRC staff agrees with this comment and has added a clarification to the description of sample size. The specific additions state that the combination of acceptable susceptibility assessments and inspection results may be used to justify not conducting an inspection at a site. However, the selection of sites used as bounding must consider all available assessments and inspection results.

Comment 8.2.8

Comment: B-4, Frequency. Add a note that this may be modified as experience and inspection data become available indicating a change is warranted in the inspection frequency.

Response: The NRC staff agrees with this comment. As with Comment 8.2.7, factors that would need to be considered are (1) operating experience at the site, (2) the susceptibility assessment for the site, (3) the results of a previous inspection(s), and (4) the susceptibility assessment and results of inspections conducted at sites that are bounding. In response to this comment, the staff divided the data collection subsection into two sections, one for canister examination and a new section for bounding analysis. The frequency remains at 5 years for either an inspection or a bounding analysis. Guidance on the documentation for a bounding analysis was added. For either specific- or general-licensed ISFSIs, a bounding analysis should include a combination of susceptibility assessments and inspections at bounding sites and previous inspection results showing no aging effects at the particular location.

Comment 8.2.9

Comment: B-5 and B-9, Elements 5 and 10. These seem like appropriate places to reference the operations-based aging management concept of analysis and periodic assessments ("toll gates") as described in NEI 14-03.

Response: As the tollgate concept in NEI 14-03 is already included in Section 3.6.1.10, the staff determined that no further changes were necessary. The staff also notes that it has not yet endorsed NEI 14 03.

Comment 8.2.10

Comment: B-6, Corrective Actions. Detection of localized corrosion (e.g. pitting) that does not exceed the acceptance criteria of IWB-3640 should not trigger sample expansion.

Response: See response to Comment 8.2.4 and 8.2.5.

Comment 8.2.11

Comment: B-7, 1st paragraph. Change "also requires an expansion" to "may warrant an expansion." The determination of the appropriate corrective actions and whether additional canisters need to be inspected should be determined as part of the corrective action and the extent of conditions evaluation that is part of the quality assurance program.

Response: The NRC staff agrees with this comment and modified the text on the extent of condition assessment accordingly. Whether an extent of condition assessment is warranted should be determined on a case by case basis. However, the assessment of whether an extent of condition assessment is or is not needed needs to be documented and justified.

Comment 8.2.12

Comment: B-7, Extent of Condition. Suggest adding reference to forthcoming EPRI susceptibility criteria. At end add "…or canisters of similar susceptibility as determined by developed criteria."

Response: The NRC staff agrees with this comment. The susceptibility assessment is under review by NRC staff, but the generic concept of a susceptibility assessment can be addressed here. NRC staff added discussion to address the susceptibility assessment and a reference to the EPRI susceptibility assessment report.

Comment 8.2.13

Comment: B-7, Disposition. Inspection results are not reportable to the NRC in accordance with 10 CFR 72.75.

Response: The NRC staff agrees with this comment and removed statements on reporting inspection results to the NRC from the Recommended Disposition of Canisters with Aging Effects discussion. See the response to Comment 8.2.2. The staff notes that 10 CFR 72.75(c)(1) states:

> (c) Non-emergency notifications: Eight-hour reports. Each licensee shall notify the NRC as soon as possible but not later than eight hours after the discovery of any of the following events or conditions involving spent fuel, HLW, or reactor-related GTCC waste: (1) A defect in any spent fuel, HLW, or reactor-related GTCC waste storage structure, system, or component that is important to safety.

Comment 8.2.14

Comment: B-7, Disposition. This section is overly prescriptive for an "example." Licensees have established corrective action programs to address this (i.e., inspection findings entered into the corrective action program, reportability is assessed, etc.).

Response: The NRC staff partially agrees with this comment. The example AMP provided is intended to serve as an example, including the actions to be taken when the acceptance criteria are not met. The NRC staff previously accepted renewal applications where specific actions were identified when acceptance criteria are not met. These included a detailed extent of condition assessment. NRC staff added a statement to the corrective actions section, indicating that alternatives to the recommended corrective actions may be provided with appropriate justification or supporting analyses.

Comment 8.2.15

Comment: B-8, Disposition. What do the alternatives to "are not permitted to remain in service" actually mean, particularly at a shutdown nuclear plant? This wording might be appropriate for class 1 piping, but is not appropriate for a low-pressure canister with no potential to create the kind of accidents we are talking about with class 1 piping.

Response: The NRC staff disagrees with this comment, and therefore, no changes were made in response to the comment. Spent fuel storage system components must either be shown to retain their design basis functions or must be repaired, replaced or mitigated to allow the design basis functions to be maintained whether the ISFSI is at a shutdown site or at an operating plant.

Comment 8.2.16

Comment: B-8, Admin Controls. Delete bullet "Frequency/methods for reporting inspection results to the NRC," as this will be covered by the industry operating experience sharing program contained in NEI 14-03.

Response: The NRC staff partially agrees with this comment. NRC staff modified the bullet to read "Methods for reporting results to the NRC per 10 CFR 72.75." The NRC also notes that the administrative controls in the example AMP should be consistent with the guidance in 3.6.1.9 of NUREG-1927.

8.3 Comments from Donna Gilmore/SanOnofreSafety.org

Comment 8.3.1

Comment: 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.

Response: The NRC staff agrees with this comment, in terms of further evaluation of a cracked canister. The continued safe operation of a canister affected by CISCC was considered in the example AMP. NRC staff notes that the referenced ASME Section XI IWB-3641(e) requires evaluation of flaws and comparison to acceptance criteria for service limits to determine the acceptability of the component for continued service. The service limits take into account loadings to which the component is subjected, including design-basis accident conditions. Therefore, the evaluation of a cracked canister (to determine if it can remain in service) would take into account the design-basis seismic event. Further information on the service conditions and service limits is provided in NRC Regulatory Guide 1.57 (ADAMS

Accession No. ML12325A043) http://pbadupws.nrc.gov/docs/ML1232/ML12325A043.pdf. Since the canister AMP already addresses this comment, the staff determined that no further changes are necessary.

Comment 8.3.2

Comment: 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

Response: The NRC staff disagrees with this comment. The example AMP is based on the requirements of ASME Section XI. As stated in response to Comment 8.3.1, the requirements of Section XI IWB-3641(e) require an evaluation to determine acceptability for continued service. There is no degradation or cracking allowed without an evaluation to determine the component is safe for continued service. Cases where a degraded component is allowed to remain in service are limited and used to coordinate repair and replacement activities, and the evaluation has to also consider the possibility for additional degradation that will occur. NRC staff note that the requirements in IWB-3641 are intended for Class 1 piping systems in operating reactors. It is also important to point out that Class 1 piping systems operate at much higher pressures than a dry storage system canister. Using an upper limit on through wall cracking developed for Class 1 piping systems is highly conservative for dry storage systems canisters. Because any incidence of cracking requires an evaluation to determine acceptability for continued service, the staff does not believe it is appropriate to specify crack percentages, and therefore, no changes were made to the guidance.

Comment 8.3.3

Comment: 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

Response: See the response to Comment 8.1.

Comment 8.3.4

Comment: 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

Response: The NRC staff notes that complex repairs of reactor components have been developed and demonstrated. Application of these methods or alternate methods to dry storage canisters requires demonstration, in compliance with the quality assurance requirements in either 10 CFR Part 72, Subpart G, or 10 CFR Part 50, Appendix B. If repair of a canister is determined to not be a feasible option, then canister replacement is an option. See the response to Comment 2.7.3. The NRC staff determined that changes were not necessary.

Comment 8.3.5

Comment: 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.

Response: It appears that the basis for this comment is, in part, incorrect information. The conditions necessary for SCC to occur have been thoroughly tested, analyzed, reviewed, and documented. An NRC staff presentation on the subject was included in the April 21, 2015, Public Meeting with the Nuclear Energy Institute on the "Chloride-Induced Stress Corrosion Cracking Regulatory Issue Resolution Protocol." The meeting summary is available on ADAMS (Accession No. ML15146A090) http://pbadupws.nrc.gov/docs/ML1514/ML15146A090.pdf and the presentation materials are available (Accession No. ML15146A115) http://pbadupws.nrc.gov/docs/ML1514/ML15146A115.pdf.

The commenter's reference is dependent on a 40-45 g/m^3 absolute humidity value in reaching conclusions regarding the temperature at which deliquescence can occur. However, the staff believes this is an unrealistic absolute humidity value, as described below, and use of this value leads to improbable assumptions regarding when deliquescence can occur, and thus improbable assumptions for crack initiation and crack growth rate. Work conducted by the NRC to assess the conditions necessary for SCC included actual records of atmospheric conditions that are available from National Oceanic and Atmospheric Administration (NOAA). In reviewing the available data from NOAA, the staff found that measured dew points above 29 degrees Celsius (84 degrees Fahrenheit) (absolute humidity of 28.8 g/m^3) are rare. According to the available records, the highest recorded dew point of 32°C (90°F) was in Appleton, WI, on July 13, 1995, corresponding to an absolute humidity of 33.7 g/m^3. The staff noted that while absolute humidity values are usually below 30 g/m^3, the conservative assessment conducted by the NRC in ADAMS Accession No. ML15146A115 used actual available "real-life" atmospheric data along with numerous conservative assumptions. The staff used these conservative assessments in formulating the inspection criteria for canisters.

NRC staff noted that the actual assessment of the conditions on the Diablo Canyon and Hope Creek canisters is 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," Charles R. Bryan and David G. Enos, July 2014 (http://prod.sandia.gov/techlib/access-control.cgi/2014/1416383.pdf). The conclusion explains

the limitations of the sample collection and analyses. The staff added the Sandia National Laboratories report to the references in the example AMP.

Comment 8.3.6

Comment: 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.

Response: See the response to Comment 8.1.

Comment 8.3.7

Comment: 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"

Response: Determination of what part of canisters are or are not accessible will depend on many parameters including details of system designs, methods, and inspection tooling. Understanding which areas of the canisters will or will not be accessible is part of ongoing efforts to develop inspection delivery systems for canister inspections. See the response to Comment 8.1. NRC staff has determined that changes are not necessary.

Comment 8.3.8

Comment: 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.

Response: Dry storage systems for spent fuel have been in use in the United States since 1986. As of December 2015, there are more than 2,200 dry storage systems in use across the country. These systems, which conform to the regulatory requirements of 10 CFR Part 72, have been used to safely store spent fuel. The development of systems for nondestructive examination of canisters will significantly improve the quality and resolution of canister examinations. Operational experience will be accumulated with time as additional examinations are conducted. The NRC may call attention to relevant operating experience as needed through generic communications. NRC staff deleted the statement on no cases of CISCC being identified, and added the Sandia National Laboratories report describing the analysis of samples collected from canisters at Hope Creek and Diablo Canyon to the references in the example AMP.

Comment 8.3.9

Comment: 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 Comments NUREG-1927 Rev. 1 6 of 7 Donna Gilmore 8/21/2015 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.

Response: The example AMP, as written, contains a publically available reference that fully describes the examination conducted at the Calvert Cliffs site-specific ISFSI as well as the examinations conducted at Hope Creek and Diablo Canyon. NRC staff added the Sandia National Laboratories report describing the analysis of samples collected from canisters at Hope Creek and Diablo Canyon to the references in the example AMP.

Comment 8.3.10

Comment: 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."

Response: The example AMP, as written, contains a publicly available reference that fully describes the examination conducted at the Calvert Cliffs site-specific ISFSI as well as the examinations conducted at Hope Creek and Diablo Canyon. NRC staff added the Sandia National Laboratories report describing the analysis of samples collected from canisters at Hope Creek and Diablo Canyon to the references in the example AMP.

Comment 8.3.11

Comment: 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.

Response: See the responses to Comments 8.1, 8.3.4, and 5.3.1.

8.4 Comments from Richard Morgal

Comment 8.4.1

Comment: Living just under 45 miles downwind from the now shuttered San Onofre Nuclear Generation Station (SONGS), I am very uncomfortable with how SCE plans on storing the plant's nuclear waste.

A majority of my concern is related to the exposure of the stainless steel dry storage canisters (DSC) to the abundant salts found in the oncoming ocean breeze used to cool these canisters. It is known by the NRC that it can take less than 16 years for a chloride induced stress corrosion crack (CISCC) to breach the canisters that SCE plans on deploying at SONGS.

Response: The NRC staff finds there is no basis to support the commenter's assertion that canisters may have through-wall cracks in as little as 16 years. The time necessary for through-wall cracks has been evaluated for hot humid environments close to coastal locations with high concentrations of sea salt aerosols, as provided in the Electrical Power Research Institute (EPRI) Flaw Growth and Flaw Tolerance Assessment for Dry Cask Storage Canisters

(EPRI, Palo Alto, CA: 2014. 3002002785). The EPRI analysis, using conservative criteria, shows that when conditions for stress corrosion cracking are present, through wall flaws may occur in 26.5 to more than 80 years depending on the ISFSI environmental conditions. The EPRI work considered multiple years of site specific weather data that is made available by the National Oceanic and Atmospheric Administration (NOAA). A summary of the results is tabulated in the referenced report in Table 3-3 on page 3-13. See the EPRI report here: http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?productId=000000003002002785.

NRC staff also evaluated conditions for cracking of dry storage canisters and found that even with conservative assumptions (i.e., conditions favorable to stress corrosion cracking initiation and propagation), the fastest crack growth rates would not result in the penetration of a dry storage canister until at least 30 years based on site specific NOAA data for 2014. The NRC assessment considered operational experience with chloride induced stress corrosion cracking that was included in Information Notice 2012-20. (http://pbadupws.nrc.gov/docs/ML1231/ML12319A440.pdf). The NRC presentation on the subject is included in the April 21, 2015, Public Meeting with the Nuclear Energy Institute on the "Chloride-Induced Stress Corrosion Cracking Regulatory Issue Resolution Protocol." The meeting summary is available on ADAMS (Accession No. ML15146A090) http://pbadupws.nrc.gov/docs/ML1514/ML15146A090.pdf and the presentation materials are available (ADAMS Accession No. ML15146A115) http://pbadupws.nrc.gov/docs/ML1514/ML15146A115.pdf.

Although there are several differences in the analysis reported by EPRI and presented by the NRC, such as parameters used to determine crack growth rates, it is important to note that both the EPRI report and the independent NRC analyses reached similar conclusions on the range of possible crack growth rates and the important influence of site specific conditions. Based on these results of known operating experience and both the EPRI and NRC independent analyses, the NRC staff finds there is no basis to support the commenter's assertion that canisters may have through wall cracks in as little as 16 years. The NRC staff has been actively involved in addressing the potential for stress corrosion cracking of welded stainless steel canisters for more than 10 years. NRC staff will continue to collect and review relevant operational experience on atmospheric exposure and stress corrosion cracking of austenitic stainless steels. If necessary, the NRC will issue generic communications to NRC licensees and CoC holders.

The NRC staff determined that changes to the guidance are not necessary.

Comment 8.4.2

Comment: This information, when coupled with the results of a rare inspection at Diablo Canyon's Nuclear Power Plant DSCs performed by EPRI in January of 2014 (which documents the presence of conditions that are necessary for a CISCC crack to initiate on canisters that have been deployed for less than two years), it seems likely that DSC's will breach long before they are relocated to a permanent repository. Some of the canisters at SONGS have been deployed for over 12 years to date, with no permanent repository planned to be available for decades.

Response: The conditions necessary for SCC to occur have been thoroughly tested, analyzed, reviewed, and documented. See the response to Comment 8.3.5 and 8.4.1. The NRC has also reviewed the available information from the inspection of canisters at Hope Creek and Diablo Canyon including the information 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," Charles R. Bryan and David G. Enos, July 2014 (http://prod.sandia.gov/techlib/access-control.cgi/2014/1416383.pdf). The NRC staff has added the Sandia National Laboratories report describing the analysis of samples collected from canisters at these locations to the references in the example AMP.

Comment 8.4.3

Comment: What is especially chilling to me and my family are public statements made by Dr. Kris Singh, CEO of Holtec, the manufacturer of the DSC system SCE would like to deploy at SONGS. During an SCE sponsored community engagement panel discussion, Dr. Singh states that one microscopic breach of the stainless steel canisters planned for SONGS would release millions of curie of radiation into the surrounding environment.

A nuclear accident of this magnitude happening so close to a major Interstate will surely force the public to rethink nuclear power and the NRC's ability to store the waste, IF a breached canister is ever detected. Which is my main difficulty with NUREG 1927 as it has been drafted.

It appears that NUREG 1927 has been written to reduce the possibility of a breached canister ever being detected. Since a canister breach releasing a radioactive plume is invisible, relatively rapid and the human biological response to radioactive exposure is long delayed, it is unlikely anybody driving on Interstate 5 will ever know of their radiation exposure from a breached DSC at SONGS.

Response: The NRC staff disagrees with this comment. Aging Management Programs that are based on inspections and monitoring are necessary when aging effects are possible on structures systems and components that are important to safety. The inspections that will be conducted as part of the aging management program are relied upon to detect aging effects, such as the potential aging of welded stainless steel canisters by localized corrosion and stress corrosion cracking. As indicated in Section 3.6 of NUREG-1927, Revision 1: An effective AMP prevents, mitigates, or detects the aging effects and provides for the prediction of the extent of the aging and timely corrective actions before there is a loss of intended function. The NRC is addressing potential storage canister cracking in the renewal period of licensed operation by requiring periodic inspections of certain canisters to provide for the early detection of cracks well before they can grow through a canister wall, with associated mitigation if necessary to maintain the system's important-to-safety functions. The dry storage system's structures, systems, and components important to safety must either be shown to continue to perform their intended functions for the requested period of extended operation or must be repaired or replaced to allow the intended functions to be maintained, whether the ISFSI is at a shutdown site or at an operating plant.

The ISFSI at San Onofre is required to be in compliance 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. The ISFSI at San Onofre has multiple Thermoluminescent dosimeters (TLDs) on the ISFSI boundary fence. These TLDs are regularly monitored. The results of the monitoring program is 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. See also the responses to

Comments 8.2.5, 8.2.15, 8.1, 8.3.4, 5.3.1. The NRC staff has determined that changes to the guidance are not necessary.

Comment 8.4.4

Comment: Please describe how a breached canister will be detected given that SCE is allowed by the NRC to remove all installed radiation detection equipment at SONGS once the plant is decommissioned. SONGS will be decommissioned for decades, with no installed radiation detection equipment for decades, before the spent nuclear fuel is relocated to a federal permanent repository. How will breached canisters be detected and how will the public be notified of a radioactive release from a CISCC breached canister? This information should be included in any comprehensive aging management plan.

Response: The NRC staff disagrees with this comment. See the response to Comment 8.4.3. ISFSI licensees (including the San Onofre ISFSI) are required to be in compliance with NRC regulations, including the radiological dose limits in 10 CFR 72.104 and 72.106. Whatever method the licensees use to demonstrate compliance with the dose limits (e.g., TLDs on the ISFSI boundary fence) must remain in use as long as the ISFSI is in operation. The NRC staff determined that changes to the guidance are not necessary.

Comment 8.4.5

Comment: How likely will a quarterly manual walk-through radiation inspection of the DSC's on-site be able to detect a breached canister? Can you please provide a probabilistic risk analysis of the likelihood that breached canisters will be detected by the quarterly manual walk-through radiation inspections.

Response: See the response to Comment 8.4.3 for (1) the purpose of an Aging Management Program to detect aging effects, (2) regulatory requirements for doses to members of the public from ISFSI operations as defined in 10 CFR 72.104 and 72.106, and (3) how to obtain NRC inspection reports that include information on the San Onofre ISFSI radiation monitoring program. NRC staff determined that changes to the guidance are not necessary.

Comment 8.4.6

Comment: Is there any data known by the NRC that determines the duration and intensity of a CISCC breached DSC's radiation release? If a microscopic CISCC breach is modeled, what means have been used to validate the model as correct? Please include this data in the NUREG 1927 so the public can be assured that the NRC has determined the likelihood of detecting a breach from the quarterly walk-through radiation inspection.

Response: See the response to Comment 8.4.3 for (1) the purpose of an Aging Management Program to detect aging effects, (2) regulatory requirements for doses to members of the public from ISFSI operations as defined in 10 CFR 72.104 and 72.106, and (3) how to obtain NRC inspection reports that include information on the San Onofre ISFSI radiation monitoring program.

Risk assessments for spent fuel storage and transportation have been conducted by the NRC and industry. These risk assessments are publically available. NRC risk assessments are published in NUREG-1140, "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees"; NUREG-1864, "A Pilot Probabilistic Risk

Assessment of a Dry Cask Storage System at a Nuclear Power Plant"; and NUREG-2125, "Spent Fuel Transportation Risk Assessment" (ADAMS Accession Nos. ML062020791, ML071340012, and ML14031A323, respectively). Of these risk assessments, the most relevant is the analysis in NUREG-1140 that analyzed a postulated accident involving the removal of the lid of a storage cask in which all the fuel rods have been damaged. The analysis found that the resultant doses were below the Environmental Protection Agency's protective action guidelines for taking protective action after an accident.

The NRC staff determined that changes to the guidance are not necessary.

Comment 8.4.7

Comment: I would also ask that the quarterly radiation inspection breach detection rate be compared to the breach detection rate using a real time 24/7 radiation monitoring system. What level of confidence would be added by relying upon a 24/7 radiation monitoring system versus a quarterly walk through inspection process?

Could there be any value to detecting a DSC breach immediately rather than months after it occurred? Could the US government be setting itself up for a huge legal battle related to negligence if hundreds of thousands of cars on Interstate 5 passing through a radiation plume without knowing? Since there would be no way of telling who drove on Interstate 5 during the breach, there would be no way to determine who's possible radiation related illness would have to be covered by the US government. Could it be that one DSC breach that went undetected for a matter of days, or months, be the lynch pin that forces all illnesses possibly contracted from exposure to radioactive krypton gas to be covered 100% by the US government? As a tax payer I question the cost savings of not monitoring all DSC's in real time 24/7 for a radiation release when it is possible that taxpayers could become financially responsible for radiation related illnesses of millions of US citizens.

Response: See the response to Comment 8.4.6.

Comment 8.4.8

Comment: If a breach is detected by a manual walk-through inspection, what would be the radiation exposure of the inspector performing the inspection? Could that level of exposure be compared to the level of exposure of the security staff that is present during repeated eight-hour shifts, day after day, unknowingly being exposed to a DSC breach that was not detected for weeks after the breach occurred? Will the ISFSI security staff be wearing personal radiation detectors? If so, should these personal radiation detectors be included in the aging management plan as early warning detectors for a breached canister? Please describe the role of personal radiation detectors for security personnel in the AMP and how they would be used to protect ISFSI employees and the general public.

Response: The Aging Management Program does not address the role of security personnel or personnel radiation detectors. See the response to Comment 8.4.6.

Comment 8.4.9

Comment: The NRC would be motivated to either set up an aging management plan that is very unlikely to detect a breached canister to avoid a public relations backlash or become hyper sensitive to radiation being released from each and every DSC. It is my opinion that the NRC

has chosen to gamble by minimizing the cost of storing the nation's spent nuclear fuel for an indeterminate amount of time, putting the taxpayer on the hook should anything go wrong. Lack of 24/7 real time radiation monitoring, coupled with the "lead" canister approach to re-licensing are the leading indicators of this gamble.

NUREG 1927 allows SCE to choose ONE canister and perform an "inspection" on that singled out canister for the first time after 20 years of being deployed in the corrosive ocean air. Then once every 5 years that same ONE singled out canister is to be "inspected" again, with each "inspection" being used to certify the integrity of the 100+ canisters storing 3½ million pounds of spent nuclear fuel on the SONGS site.

Response: The NUREG provides guidance on license and CoC renewals as part of the NRC's mandate to protect the public and the environment from the hazards of radioactive materials. The example AMP provides one acceptable method for canister examination that identifies the canister most susceptible to corrosion. The assessment may require additional canisters to be examined. See the responses to Comments 8.4.3 and 8.4.6.

Comment 8.4.10

Comment: Have assessments been performed on the uniformity of CISCC growth on all stainless steel DSCs that experience the exact same conditions? Can we be assured that all the conditions are exactly the same for all the DSCs at a given ISFSI? Will edge DSC's be cooler than DSC's residing in the middle of the ISFSI field? How can we be sure that different DSC's don't behave differently when one of the initiators of CISCC is stress that occurred during the manufacturing process? How can ONE canister be used to certify a field of 100+ DSCs that have each been manufactured slightly differently?

Response: The staff provides the following information to respond to the questions posed in the comment, but determined that no changes to the guidance were needed.

See the response to Comment 8.4.1.

EPRI Report No. 3002005371, "Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless Steel Canisters for Dry Cask Storage Systems," further presents a set of assessment criteria for evaluating individual independent spent fuel storage installations (ISFSIs) containing welded stainless steel canisters for their relative susceptibility to chloride-induced stress corrosion cracking (CISCC). These criteria were developed in a manner that provides a ready ranking of a given site so that focus and effort can be prioritized at geographic locations where ISFSIs may be more susceptible to CISCC. The ISFSI ranking criteria considered the environmental factors that have the greatest influence on CISCC degradation. Additionally, canister ranking criteria were devised to provide prioritization of canisters at a given ISFSI for actions to address CISCC susceptibility. Guidance is also provided for determining when a given canister may be considered bounding of another canister, including at a different ISFSI, for CISCC susceptibility.

Section 3.4.1.4, which is incorporated by reference in Section 3.6.1.10, states that consideration should also be given to event-driven fabrication or operational issues that may contribute to degradation when selecting SSCs for inspection (e.g., welding repairs, occurrence of natural or man-induced events, exposure to potentially corrosive environments prior to the storage term).

In response to Comment 8.2.4, the staff has added clarification to the AMP on acceptance criteria for visual inspections. Consistent with the licensee's approved Quality Assurance Program, the licensee is expected to perform a root cause analysis and verify the extent of condition (i.e., determine if additional systems need to be inspected), as appropriate, for any conditions identified in the inspection that could result in a loss of intended function (see Section 3.6.1.7, "Corrective Actions"). The extent-of-condition assessment may require additional canisters to be examined.

The NRC recognizes the need to optimize inspections, while also considering occupational doses and maintaining doses as low as is reasonably achievable (ALARA).

Comment 8.4.11

Comment: Then there is the possibility of insects being attracted to the warm DSC vent holes with prevailing winds creating a selection process that is not uniform for the likelihood of insects going into the ventilation openings of the concrete over pack. Insect urine and blood are known to initiate CISCC and with a higher incidence of edge DSC's possibly getting higher insect infiltrations there may be variation across the ISFSI field that are not accounted for in the lead canister selection process. Have these possible variations been considered before the NRC and industry selected the lead canister process?

Response: EPRI Report 3002005371, "Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless Steel Canisters for Dry Storage Systems," addresses considerations for the selection of canisters for inspections. Stainless steels were first patented in 1912 and were in widespread use by the 1930s. The NRC has reviewed operational experience with the use of stainless steels in industrial, commercial and civil engineering applications and concluded that no operational experience to date indicates degradation of stainless steels as a result of either insect or microbial activity for any application that is similar to conditions of dry storage. Nevertheless, the NRC staff will continue to review relevant operational experience and will address issues deemed significant for welded stainless steel dry storage canisters through the use of generic communications or by other means available to the NRC. The NRC staff determined that changes to the guidance are not necessary.

Comment 8.4.12

Comment: Chloride Induced Stress Corrosion Cracking (CISCC) has been know for nearly 100 years and has been used by processing plant engineers for decades due to stainless steel's tendency to leak long before it ever breaks outright. Thus providing plant maintenance personnel a margin of error where a liquid filled stainless steel tank will report its breach, by leaking, well before it ever bursts. But with gaseous radioactive contents this feature of stainless steel should be considered a danger to public health and safety.

Yet CISCC in stainless steel has proven very difficult to detect before a breach due to its microscopic surface area, while penetrating deeply into the stainless steel. Given the enormity of the eight-foot diameter, eighteen feet long, 100-ton canisters it has never been demonstrated that microscopic CISCC penetrations have ever been reliably detected on a DCS containing spent nuclear fuel. Once again I rely upon Dr. Kris Singh stating his concerns about how difficult, if not impossible it is to detect a CISCC crack on one of his company's stainless steel DCSs. Please publish your latest results of the nuclear industry's development to detect and quantify CISCC damage to stainless steel DSCs so we can all rest assured that the problem is

solved. Because right now it seems like we are relying upon non-existent technology to keep us from experiencing radioactive harm with no plan other than to just wait.

Response: Stainless steels were first patented in 1912 and extensively developed by the mid-1930s. Stress corrosion cracking was observed in the 1940s but not understood. Extensive research and testing on stress corrosion cracking of stainless steels was conducted starting in the 1970s. With respect to inspection and detection of SCC, see the response to Comments 8.1, 8.3.4, and 5.3.1.

Comment 8.4.13

Comment: To date, at least in the NRC public literature, not one loaded canister has ever been removed from its concrete over pack to be inspected for CISCC. All DSC inspections that have been made for ISFSI re-licensing have been performed through very small ventilation openings in the concrete over pack with a very small percentage of the overall canister area being inspected. Why has there been no minimum percent of the canister's surface area been established for inspection of the DSCs? So the ONE canister that will be inspected to re-license the entire ISFSI will have only a small percentage of its surface area inspected (with no stated minimum surface area to be inspected)?

How can a quality control inspector, evaluating the thoroughness of a DSC inspection at an ISFSI re-licensing inspection, determine that sufficient inspection of the lead DSC has occurred to allow the ISFSI to be re-licensed if there is no stated minimum DSC surface area to be inspected during the inspection? To be a valid inspection there needs to be clearly specified a minimum percent of the DSC's surface area to be inspected to ensure that a valid inspection has occurred. Please include a minimum surface area of the lead DSC that will be inspected to enable an ISFSI to be re-licensed.

Response: See the response to Comments 8.1 and 8.3.7, and 8.4.10.

Comment 8.4.14

Comment: Given the microscopic nature of CISCC, it is extremely unlikely that a visual inspection will detect a CISCC crack. How can NUREG 1927 be accepted by the public when there are no minimum DSC inspection area specifications, no CISCC blemish size or penetration specifications, no equipment specifications to perform the inspections and no procedures stated that can be used to quantify the time predicted until a given CISCC blemish can no longer be relied upon before a breach will occur? Please include measurable inspection criteria to NUREG 1927 or call out the NRC documents that specify the measurable inspection criteria to allow an ISFSI to be re-licensed, as there is not sufficient measurable inspection criteria stated in the current draft of NUREG 1927.

Response: With respect to inspection and detection of SCC, see the response to Comments 8.3.3, 8.3.4, 5.3.1, and 8.3.7. Note that the requirements for detection are included in the inspection methods identified in the ASME code.

Comment 8.4.15

Comment: I have to ask about how DSC's exposed to the ocean's salt air will be treated when stored in a high seismic area? How will the CISCC damage be detected and quantified to ensure that there aren't one hundred plus salt corroded DSC's that breach all at once during a

strong earthquake event? How can the NRC ensure the public that this will not happen when all seismic certifications/simulations for DSC's are performed upon brand new DSC's with no flaws? Please provide some sense of how the NRC intends to quantify CISCC damage to aged DSCs that reside in high seismic areas when there is no way of detecting CISCC damage?

Assuming industry is able to detect microscopic CISCC cracks with 100% accuracy, what will be considered an acceptable depth of a crack to allow an aged canister to reside in a high seismic area? The 75% through-wall crack suggested by ASME is not for a vessel containing radioactive material and is not specified for high seismic areas. It is hard for an engineer to comprehend how such a situation can be confidently managed amongst the eight million people that live within a 50-mile radius of SONGS. Dr. Kris Singh's millions of curie for one canister could be 100's of millions of curie of radiation with 100+ canisters breaching due to a strong seismic event. The NRC can not justify looking at CISCC and high seismic vibrations as separate, unrelated conditions, they need to be evaluated together for locations near the ocean and fault lines.

Response: See the response to Comment 8.3.1.

Comment 8.4.16

Comment: The "lead" canister approach to certification of the entire field of 100+ canisters at any Independent Spent Fuel Storage Installation (ISFSI) demonstrates how unwilling the NRC and SCE are towards doing anything to the canisters, except let them sit. Without real time radiation detection, without meaningful inspection I have to ask, what would be done if somehow it is determined that a CISCC breach did occur?

Please state where the NRC documents what will be done when a DSC has aged to the point of breaching from CISCC. It is likely this will happen and it should either be in the aging management plan (AMP) or the AMP should reference the NRC document(s) that describes in detail what will be done, what equipment is needed on-site, and how long the DSC repair is to be relied upon before it requires further inspection. This information needs to be documented before DSCs are deployed on-site to ensure equipment needed will be at hand to repair a breached DSC. In addition, once one DSC has breached what will be the procedure to evaluate the remaining aged DSC's at the ISFSI site?

Response: See the response to Comment 8.4.3 for the purpose of an Aging Management Program to detect and manage aging effects. In general, AMPs require corrective actions when acceptance criteria are not met. AMPs are developed by either specific licensees or CoC holders and submitted to the NRC in specific license or CoC renewal applications. The corrective actions must be sufficient to maintain the safety function of the structures, systems, and components that are important to safety. Specific actions to be taken if a welded stainless steel canister is affected by CISCC will be provided by the site specific licensees or CoC holders in renewal applications. NRC staff determined that changes to the guidance are not necessary.

Comment 8.4.17

Comment: I would think that the AMP would include all aspects of aging canisters including transportation of aged canisters. What equipment will be required to ensure that canisters that have been exposed to the ocean's salty air for decades will be robust enough to transport? How will these aged canisters be reliably inspected given the lack of any demonstrated ability to inspect for CISCC? Assuming industry is able to detect microscopic CISCC cracks with

100% accuracy, what will be considered an acceptable depth of a crack to allow an aged canister to be transported? What criteria will be used to evaluate this allowance? Please relate the transportation CISCC crack penetrations maximums to high seismic area calculations to provide a sense of how robust a transport cask must be in relation to a cask that is expected to survive a strong earth quake event. If a crack is determined to be too deep, what is the procedure to repair or replace the damaged canister? The equipment required also needs to be included in an aging management plan. Please specify.

Response: See the response to Comment 8.4.3 for the purpose of an Aging Management Program to detect and manage aging effects. The staff notes that NUREG-1927 is only applicable to storage renewal applications, and does not address transportation of spent fuel. A canister would need to meet the criteria of the transport CoC before it could be transported. See the response to Comment 2.2.4 regarding transportation.

Comment 8.4.18

Comment: It appears that NU-REG 1927 has been set up to minimize the likelihood of a CISCC breached canister ever being detected. I would argue that the cost of 24/7 real time monitoring and Internet reporting of DSC radiation retention would provide the public with hard evidence that there are no current radiation leaks, and over the test of time that stainless steel DSCs are robust and long lasting. If the industry and NRC are so confident that this storage technology is robust enough to endure the high salt content found in the ocean air, there is little reason to avoid a robust radiation monitoring approach. A rigorous radiation monitoring approach provides confidence to other on-site ISFSI environments that experience less harsh exposure to salts than SONGS, that these less harsh sites are safe as well. Real time radiation release information immediately following an earthquake could provide the general public insight into how to respond to a possible rapid release of radioactive gas emitted from an aging ISFSI on an earthquake fault. If no means to combine both CISCC and Seismic impacts can be derived, maybe 24/7 radiation monitoring and reporting could be the mitigating solution to this complex problem. The reduced liability of stopping people from exposing themselves to a radioactive plume on Interstate 5 might be enough of a public safety benefit to consider SONGS to become the “DSC robustness demonstration” venue with 24/7 radiation monitoring and reporting. But NUREG 1927 leaves the public in the dark with little to no confidence that the NRC is doing anything more than hiding the elephant in the room. The public deserves better, proving the spent nuclear fuel storage technology’s ability to reliably contain it contents even in the harshest environment found at SONGS with radiation monitoring equipment and reporting. The public is hoping it is going to work and should be immediately notified if the canisters fail.

Response: As this comment is a summary of the commenter’s previous comments, please see responses to Comments 8.4.1–8.4.17.

8.5 Comments from Ray Lutz/Citizens’ Oversight Projects

Comment 8.5.1

Comment: 3. CORRECTIVE ACTIONS MISSING: The sample AMP provided in Appendix B has a section called “Corrective Actions” on page B-6. But there this section only covers changes in surveillance frequencies and sample sizes. It unfortunately does not cover actual corrective actions.

On page B-7, the following sentence is the closest we get to any specification of corrective actions.

Canisters with confirmed localized corrosion or stress corrosion cracking must be evaluated for continued service. Canisters with localized corrosion or stress corrosion cracking that do not meet the prescribed evaluation criteria must be repaired or replaced.

> What is the procedure(s) used for repair?> What is the procedure(s) used for replacement?

Each of these should be specified as referenced documents.

The aging management program with respect to spent fuel canisters includes only half of what any program should provide. Sure, we must monitor the canisters with regard to aging. But once cracking or other degradation exceeds threshold criteria, a method to correct the aging is required. Thus, it is necessary to specify procedures which will be used for repair or replacement of the canisters.

If the canisters are to be "replaced," without more detail in terms of a document specifying how this will be done may mean that spent fuel storage sites will not be prepared to deal with a repair or replacement.

Response: The NRC does not prescribe how licensees would take corrective action, rather 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. See also the response to Comments 8.3.4 and 8.2.1.

Comment 8.5.2

Comment: Additionally, once a canister is repaired, what changes in the management program are then put into place to deal with the repair? There will likely need to be a whole new branch of the flowchart for repaired canisters.

Response: AMP Element 8, Confirmation Process, is intended to verify that appropriate corrective actions have been completed and are effective, and includes provisions to preclude repetition of significant conditions adverse to quality. As guidance on the confirmation process is already included in Section 3.6.1.8, NRC staff determined that changes are not necessary.

CHAPTER 9: RESPONSES TO PUBLIC COMMENTS ON EXAMPLE AMP FOR REINFORCED CONCRETE STRUCTURES

The comments on Example AMP for Reinforced Concrete Structures relate to Appendix B, Table B-2 of NUREG-1927, Revision 1.

9.1 Comments from Robert Einziger/NWTRB

Comment 9.1.1

Comment: P-B16, - Define "finding".

Response: The NRC staff changed the term "finding" to "condition," to reflect information or observations identified in a licensee's inspections, where appropriate.

Comment 9.1.2

Comment: P-B13, - Terms such as "spalling", "scaling", "curling", "deflections", "honeycombing", and "popouts" should be defined. Suggest doing it in a footnote. "Adequate" and "heavy" are unquantifiable terms. Either define or change wording.

Response: The referenced terminology is consistent with terminology in American Concrete Institute Standard CT-13, "ACI Concrete Terminology". The NRC staff added a reference to this standard. In addition, the staff removed terms "heavy" and "adequate" where appropriate.

9.2 Comments from Nuclear Energy Institute

Comment 9.2.1

Comment: B-12, 16-18 (and in each element of the AMP in Table B-2). AMP indicates use of radiation surveys to ensure shielding function. Radiation surveys are routinely performed by the licensees to ensure compliance with 10 CFR 72.104. Use of radiation surveys results cannot be a CoC holder activity.

Response: The NRC staff agrees with the comment and provided clarifications in Element 1 of the AMP, "Scope". The staff clarified the intent of the radiation surveys is to (1) ensure compliance with 10 CFR 72.104 dose equivalent requirements beyond the controlled area during normal and off-normal conditions of storage, and (2) monitor performance of the concrete as a neutron/gamma shield at near system locations as an indicator of concrete degradation. The staff notes that AMP activities are performed by licensees, while the CoC holder is responsible for defining those activities in the CoC renewal application.

Comment 9.2.2

Comment: B-12, 16-18. Why include activities in an aging management AMP that are already a part of the existing programmatic requirements of the DSS? These requirements would not be expected to change during the period of extended operation.

Response: Aging management programs may include activities performed during the initial storage period if these activities are relied upon for ensuring aging effects will not result in a loss of intended function of the dry storage system. The staff has determined that changes to the guidance are not necessary.

Comment 9.2.3

Comment: B-12, 18. For daily inspections, add "if applicable" at the end. Not all licenses are required to inspect the vents.

Response: The NRC staff agrees with the comment and has made the suggested change.

Comment 9.2.4

Comment: B-12, Scope, Item 2. Change "Ground water chemistry program to manage" to "Ground water monitoring program to identify." The purpose of a groundwater chemistry program is to identify aging mechanisms versus managing the groundwater chemistry.

Response: The NRC staff agrees with the comment and changed the language to "Groundwater chemistry monitoring program to identify conditions conducive to below-grade (underground) aging mechanisms…."

Comment 9.2.5

Comment: B-12, #3 2nd bullet. "as a gamma and neutron shield." Also text seems missing: what is meant by "at near system locations"?

Response: See the response to Comment 9.2.1. The determination of near-system locations is defined by the applicant and is primarily based on accessibility to the system. Element 4, "Detection of Aging Effects," has been revised to add clarity for defining the sample size and specific locations for performing AMP activities (e.g., radiation surveys).

Comment 9.2.6

Comment: B-12, Scope, Item 3. This is something licensees and CoC holders already are required to do as a part of their licenses/CoCs and is not something that really addresses aging of reinforced concrete structures. Recommend removing Item 3.

Response: See the response to Comment 9.2.2.

Comment 9.2.7

Comment: B-12, Scope. Change "provides a means to address" to "provides a means to identify."

Response: The staff changed the text, "The scope of the program provides means to address" to "The program provides means to address" to refer generally to the aging management program. The aging management program includes 10 elements, each of which provides details for addressing aging effects, including preventive actions, parameters monitored, detection of aging effects, acceptance criteria, monitoring and trending as well as corrective actions and other quality assurance requirements. Therefore, the AMP is not only written to

ensure detection, but also ensure corrective actions are taken, as appropriate, for conditions not meeting established acceptance criteria.

Comment 9.2.8

Comment: B-13, TLAA criteria for concrete neutron fluence. There is little to no change in mechanical properties for neutron fluence up to 10^19 n/cm^2 (10^23 n/m^2).

Response: The staff added footnote to clarify that "The staff recognizes the critical cumulative fluence in ACI 349.3R-02 as adequately conservative. The applicant may choose to justify a higher critical cumulative fluence if a technical bases is provided."

Comment 9.2.9

Comment: B-13, Scope-Fluence Numbers. These are quite a bit lower than the values in NUREG/CR-7153, Vol. 4 (i.e., 10^19n/cm^2).

Response: See the response to Comment 9.2.8.

Comment 9.2.10

Comment: B-13, Scope-temperature Limits. Temperature can be as high as 350°F for short-term events including off-normal events, per ACI and many DCS licensing bases. This should be reflected in the guidance.

Response: The staff reviewed the comment and has revised text to remove the TLAA approach for high-temperature dehydration of the concrete. The staff recognizes acceptable higher temperature criteria (alternative to the temperature requirements of ACI 349) during the review of the initial design bases (see NUREG-1536, Revision 1). Therefore, consistent with the approved design bases, aging effects due to high-temperature dehydration of the concrete are not considered significant.

Comment 9.2.11

Comment: B-13, Preventative Actions. In addition to inspection of vents, add "and/or temperature monitoring, as applicable."

Response: The staff agrees with the comment and has made the suggested change.

Comment 9.2.12

Comment: B-13, Scope-Last sentence. Protective coatings on painted steel need to be the subject of a separate AMP. It is a complex subject, and not all painted steel in the various DCS systems have similar design and safety functions.

Response: Element 1, "Scope," of the aging management program for reinforced concrete structures already reflects this comment. Therefore, the staff has determined that changes to the guidance are not necessary.

Comment 9.2.13

Comment: B-14, Parameters-"Efflorescence". This is something of a special case. This phenomenon should be monitored and extreme cases should be addressed, but it is not usually detrimental (see NUREG/CR-7153, for example).

Response: Efflorescence is a parameter monitored in the AMP in order to address potential degradation due to leaching of calcium hydroxide, per the acceptance criteria in ACI 349.3R-02. The staff determined that changes to the guidance are not necessary.

Comment 9.2.14

Comment: B-14, Parameters. "The Parameters evaluated are adequate…" — what does this sentence mean in the context of an AMP?

Response: The staff revised the statement to "The parameters consider any surface geometries that may support water ponding and potentially increase the rate of degradation."

Comment 9.2.15

Comment: B-14, Parameters. The last sentence in this section: The radiation surveys are as prescribed by each licensee's radiation protection program and therefore should not be a part of the AMP.

Response: See the response to Comment 9.2.1 and 9.2.2.

Comment 9.2.16

Comment: B-15, Detection. Crack-depth determination is generally not possible and not a good attribute to use as an example.

Response: The staff agrees with the comment and has made clarifying changes to remove "depth" and revise example attributes include "width" and "extent."

Comment 9.2.17

Comment: B-15, Monitoring and Trending. Of the examples cited, dose rate is about the only one that is readily achievable and based on field surveys and rarely changes (in other words, not a very useful parameter for trending). Photographic evidence provides the best resource to monitor change.

Response: As the AMP states, these are examples of considerations for monitoring and trending. However, the staff made revisions to the text to remove corrosion rates, as these may require non-destructive testing to determine. The staff revised other examples to include (1) crack growth/extent, (2) pore/void density and affected areas, or (3) dose rates.

Comment 9.2.18

Comment: B-16, Corrective Actions, last bullet. Change "e.g. results in the loss of intended function" to "per 72.75."

Response: The staff agrees with the comment and has made the suggested change.

9.3 Comment from Donna Gilmore/SanOnofreSafety.org

Comment: 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

Response: The staff disagrees with the assertion that inspections and repairs, as necessary, cannot be performed on concrete structures for dry storage systems to ensure their intended functions are adequately maintained. Operating experience supports the NRC's position that inspections and adequate repairs, as necessary, can be performed, as demonstrated by activities during the initial license period; for example, activities performed at the Three Mile Island Unit 2 at Idaho National Laboratory ISFSI, the Palisades ISFSI and the Calvert Cliffs ISFSI (ADAMS Accession Nos. ML12320A697, ML11097A028, ML13050A323, ML13119A242, ML13119A243, ML13119A244).

As the comment denotes, the staff held a workshop on age-related degradation of concrete structures used in dry storage systems from February 24-25, 2015. The staff notes that the workshop included discussion on potential and operable degradation modes identified in the aging managing program (AMP) for reinforced concrete structures in Appendix B of NUREG-1927, Rev. 1. The expert panel did not identify any specific technical gaps that would prevent timely detection of aging effects that could compromise the intended functions of the concrete structure during a first renewal period (up to 40 years) (see "Expert Panel Workshop on Concrete Degradation in Spent Nuclear Fuel Dry Cask Storage Systems—Summary Report" at ADAMS Accession No. ML16103A218).

The staff disagrees with the assertion that tools and methods for inspection and repair of concrete structures are not available, as the cited operating experience above has demonstrated.

The staff disagrees with the assertion that the acceptance criteria in the AMP for reinforced concrete structures in Appendix B of NUREG-1927, Rev. 1 is inadequate. The NRC has chosen to use consensus guide ACI 349-3R-02 for establishing acceptance criteria and frequency of inspections incorporated in the AMP in NUREG-1927, Rev. 1. ACI 349.3R was written by consensus between experts in concrete degradation (i.e., the guide was developed and revised by the ACI 349 committee, which involves experts from academia, industry, and the NRC. Therefore, for aging management of concrete structures in dry storage systems, the staff has chosen to use the same standards as those used for safety-related structures in nuclear power plants (see NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report"), which perform containment of active structures under more aggressive environments.

The staff notes that the guidance in NUREG-1927 is generic, and it does not discuss the specific design bases for the Holtec HI-STORM UMAX system, which was approved in April 2015. The technical bases for that renewal will be addressed when and if a renewal application is submitted for the system. The staff notes that any proposed concrete AMP for that system would need to address detection of aging effects, including the methods or techniques for inspection.

The staff determined that changes to the guidance are not necessary.

9.4 Comment from Patricia Borchmann

Comment: Appendix B – Example Aging Management Plans (AMPs). In the Final NUREG 1927 (Rev. 2), stakeholders feel that NRC also needs to expand the listed AMPs, to also examine and define a plan for a new cause of accelerated Concrete Degradation patterns, which were just recently identified this summer 2015 during an inspection at Seabrook nuclear power plant. I believe that over 8.4 million living within 50 miles of San Onofre will probably also exposed to the same type of Alkali-silica reaction (ACR) or concrete degradation, which is also commonly found in concrete structures like bridges and dams. Stakeholders in California expect the existing NUHOMS 27 dry casks inside the substantial concrete containment structure onsite at San Onofre, as well as the proposed large concrete containment system proposed by SCE Edison plans to use with the HOLTEC UMAX dry cask storage may also be vulnerable to this new type of accelerated concrete degradation.

Response: The staff clarifies that concrete structures generally do not perform a confinement safety function in dry storage systems.

The staff recognizes that additional example aging management programs (AMPs) would be helpful to licensees and CoC holders in the preparation of their renewal applications. The staff will include additional AMPs in the more extensive Managing Aging Processes in Storage (MAPS) report, which will be provide additional information for aging management of dry storage systems (similarly to the Generic Aging Lessons Learned Report (NUREG-1801) used for reactor license renewal). The MAPS report is expected to be issued for public comment in 2016.

The staff recognizes that the concern of potential alkali-silica reaction (ASR) degradation is of importance to San Onofre residents. Although, to date, no operating experience on ASR has been reported in any concrete structure used in dry storage systems, the staff recognizes the potential for ASR during extended periods of operation for the reasons identified in Information Notice (IN) 2011-20 (ADAMS Accession No. ML112241029). Therefore, the example AMP for reinforced concrete structures included in Appendix B assumes ASR as a potential operable degradation mode and includes activities for ensuring that ASR, if it occurs, is timely detected and addressed to ensure no loss of intended function for the concrete structure.

The staff has determined that changes to the guidance are not necessary.

CHAPTER 10: RESPONSES TO PUBLIC COMMENTS ON EXAMPLE OF A HIGH BURNUP FUEL MONITORING AND ASSESSMENT PROGRAM

The comments on Example of a High Burnup Fuel Monitoring and Assessment Program relate to Appendix B, Table B-3 of NUREG-1927, Revision 1.

10.1 Comments from Robert Einziger/NWTRB

Comment 10.1.1

Comment: P-B20, 4 th paragraph - The guidance in ISG-24, and Appendix D gives the applicant a way to use the results of a demonstration if the boundary conditions on burnup, temperature, and cladding type, of the demonstration are not met. Add a statement to this effect in the AMP.

Response: The staff agrees with the comment and has added a footnote to Element 1, "Scope of the Program" to address the comment.

Comment 10.1.2

Comment: P=B25 - item #2 after "surrogate experiment" -Add "of a duration similar to that specified in Appendix D".

Response: The guidance is clear that separate effects surrogate experiments should provide reasonable assurance that the acceptance criteria in Element 6 of the AMP will continue to be met during the renewed storage period. The staff considers that this is ultimately the criteria by which the assessment will be reviewed. The staff determined that changes to the guidance are not necessary.

10.2 Comments from Nuclear Energy Institute

Comment 10.2.1

Comment: B-20, Middle of page. The HDRP is now using maximum assembly average burnup 55.5 GWD/MTU. (ML15133A082)

Response: The staff revised the description of the HDRP and added a reference for the burnup to address the comment.

Comment 10.2.2

Comment: B-20, Last paragraph in 1. Burnup should be clarified as "rod-average" burnup, when discussing burnup. Maximum rod-average burnup for commercial licensees is 62 GWD/MTU.

Response: The staff agrees that "burnup" should be clarified. However, the staff disagrees with the use of the "rod-average burnup" term in this area. The staff believes that "assembly-average burnup" is a more appropriate term, as this is the term typically used in

licenses and CoCs for the specifications of the fuel that may be stored. The staff changed the guidance to indicate "assembly-average burnup."

Comment 10.2.3

Comment: B-20, Last paragraph in 1. The burnup, cladding type and temperatures of any demonstration program should not be used to limit the applicability of a demonstration program. The original low burnup demonstration project conducted at Idaho National Lab project covered only Zircaloy-4, with assembly average burnups of approximately 36 GWD/MTU, but the NRC extended applicability up to 45 GWD/MTU and all zirconium-based alloys. Individual licensees should be able to make a case as to the applicability of a demonstration program to the fuel they have stored at their ISFSI.

Response: The staff agrees with the comment and has made a change to the text to clarify. See the response to Comment 10.1.1.

Comment 10.2.4

Comment: B-20, Last sentence in 1 (#3). Use of design basis temperatures to compare to the best-estimate demonstration cask temperatures is not a valid comparison. Temperature comparisons between a demonstration project and loaded systems should be the same basis (i.e., best-estimate to best-estimate, or design-basis to design-basis).

Response: The staff agrees with the comment and has made a change to the text to clarify. See the response to Comment 10.1.1.

Comment 10.2.5

Comment: B-22, Footnote 2. Is the 0.43 gram-mole a new limit on moisture content for drying? New requirements should not be specified in guidance.

Response: The staff modified the text to clarify that the cited moisture content is an example and not a new limit. The footnote adequately states the pertinent reference (NUREG-1536, Revision 1), which discusses the expected water content of 0.43 gram-mole for a given set of drying conditions (which the application identifies if it is pertinent to the design bases subject to renewal).

Comment 10.2.6

Comment: B-22, Footnote 3. Suggest deleting. Does not add value.

Response: The staff agrees with the comment and has made the suggested change.

Comment 10.2.7

Comment: B-25, #2, last sentence. The risk of opening a cask/canister should be evaluated against the risk of cladding degradation occurring in a cask/canister meeting it's design basis. Does this requirement to open a cask at each site make regulatory sense from a risk perspective?

Response: The staff notes that example AMPs in Appendix B do not impose new regulatory requirements. The example AMPs provide staff-accepted approaches for addressing aging effects of structures, systems, and components within the scope of license or CoC renewal, to ensure compliance with the requirements in 10 CFR 72.42(a) and 72.240(c). An applicant may choose to present alternate approaches for providing confirmation that the high burnup fuel remains in an analyzed configuration during the period of extended operation. The staff recognizes that opening a cask/canister is an onerous high-risk activity. Therefore, in the example High Burnup Fuel Monitoring and Assessment Program, the staff has provided an alternate approach to opening a cask/canister in the event that destructive-evaluation data from a surrogate demonstration program has not been obtained by the date specified in formal evaluation 2 (Table B-4). This alternate approach involves providing evidence to the NRC, through separate effects surrogate experiments, that the acceptance criteria in Element 6 will continue to be met during the renewed storage period.

The staff determined that changes to the guidance are not necessary.

Comment 10.2.8

Comment: B-25, Table B-4. Table B-4 is unique relative to the other two sample AMPs. It basically provides some fallback position in the event the HBU demo program has not provided necessary data prior to those early loaded site where the initial storage term will expire prior to when demo results will be available. One of the fallback options is to initiate a corrective action. There is no need to provide further guidance as to what this needs to be. For the most part this would likely just be a long term tracking item to wait for the demo program surrogate data to be available since this may not be something that can be developed from analytical tools.

Response: See the response to Comment 10.2.7. The staff recognizes that the licensee's corrective action program (CAP) will respond accordingly to conditions that may compromise the intended function of the structures, systems, and components within the scope of the aging management programs, per the quality assurance requirements in 10 CFR Part 72, Subpart G, and 10 CFR Part 50, Appendix B. The staff determined that changes to the guidance are not necessary.

10.3 Comment from Donna Gilmore/SanOnofreSafety.org

Comment: 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."

Response: The staff disagrees with the comment.

The example of a High Burnup Fuel Monitoring and Assessment Program (AMP) was developed using the guidance in Appendix D (which is equivalent to that presented in Interim Staff Guidance (ISG)-24). ISG-24 has been informed by research on high-burnup fuel beyond the cited references in that document, including those mentioned in this comment.

The staff has extensively reviewed the work conducted by Argonne National Laboratory, and it notes that this work was partially funded by the NRC in its efforts to better understand the operational parameters resulting in radial hydride reorientation for various cladding alloys. The staff further notes that these references conclude that both the susceptibility to radial hydride reorientation and the susceptibility to embrittlement vary depending the cladding alloy. The staff, however, clarifies that these reports do not discuss the rate of degradation (kinetics), as the comment suggests. The staff recognizes that research in radial-hydride reorientation is ongoing, and therefore has issued considerations for addressing the potential issue of radial-hydride reorientation in a draft Regulatory Information Summary (RIS), "Considerations in Licensing High Burnup Spent Fuel in Dry Storage and Transportation" (ADAMS Accession No. ML14175A203). Consistent with NUREG-1927, Revision 1, the draft RIS endorses the use of a demonstration program in accordance with ISG-24 (i.e., Appendix D in NUREG-1927, Revision 1) as a conservative approach that provides confirmation on compliance with 10 CFR Part 72 regulations during normal and off-normal conditions of storage.

10.4 Comment from Patricia Borchmann

Comment: Because the Standard Review Plan proposed for NUREG 1927 (Rev. 1) will specifically apply to both low burnup, and high burnup spent fuel, the level of detail and technical degree of detail and specificity should correspond to maximum amount possible for each type of spent fuel. Stakeholders point out that there are a number of previously unexamined characteristics identified in a Memo dated December 17, 2013 by Bob Alvarez, that will need to be examined carefully, and integrated into the Final NUREG 1927 (Revision 2) pertaining to high burnup spent fuel.

A single example of high burnup spent fuel characteristics that needs additional technical analysis and thermal hydraulic study is the effect of high burnup fuel assemblies having been stored a longer time in spent fuel pools to achieve specific targeted cooling, and thermal range, would be to determine if the effect of having boron additives combined in water contained in

spent fuel pools, might cause some unintended, or unanticipated, unexpected consequence on behavior of high burnup spent fuel. Stakeholders point out that on several pages of the 4-page Memo by Bob Alvarez (dated December 17, 2013), titled "High Burnup Spent Power Reactor Fuel" there are several obvious safety deficiencies caused by NRC's decision during 1990's to allow Licensees to operate reactors longer between refueling outages, and thereby decreasing number and frequency of required fuel outages. Because the behavior of High Burnup Fuel has yet to be fully characterized, tested, and fully understood, the NRC needs to fully integrate the analysis/findings contained in the Alvarez Memo on High Burnup Fuel into the NUREG 1927 revised, or Final Revision (2).

Response: NUREG-1927, Revision 1, does not revise the staff's determination that high burnup fuel may be adequately stored in compliance with 10 CFR Part 72 regulations. Instead, NUREG-1927, Revision 1 provides guidance for ensuring that renewal applications address aging effects and ensure that the design bases of approved designs for dry storage systems and specific-licensed ISFSIs continues to be maintained in the period of extended operation.

The staff notes that the review of both low-burnup fuel (i.e., less than 45 GWd/MTU) and high-burnup fuel (greater than or equal to 45 GWd/MTU) is performed with equal rigor to ensure that each spent fuel storage cask design and ISFSI is able to meet the regulations of 10 CFR Part 72. The staff has addressed the considerations for scoping and aging management for both burnup ranges in Sections 2.4.2.1, "Scoping of Fuel Assemblies," and 3.4.1.4, "Aging Management Review of Fuel Assemblies".

NUREG-1927, Revision 1 adequately cites the references used in the development of the guidance related to high-burnup fuel, particularly for Chapters 2 and 3, and Appendices B and D. The staff continues to monitor new research on high-burnup fuel, and the NRC continues to sponsor new research to expand the technical bases supporting safe storage of high-burnup fuel. For example, the staff has recently published work performed by Oak Ridge National Laboratory (ORNL) on mechanical fatigue testing of high-burnup fuel (NUREG/CR-7198, "Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications," May 2015, ADAMS Accession No. ML15139A389). Additional work is being performed by ORNL for mechanical fatigue testing of high-burnup fuel subjected to radial-hydride reorientation, which results are expected to be published in 2016. The staff also actively participates in forums and committees where experts on fuel performance discuss arising needs for expanding the technical bases supporting safe storage of high-burnup fuel.

The staff determined that changes to the guidance are not necessary.

CHAPTER 11: RESPONSES TO PUBLIC COMMENTS ON LEAD SYSTEM INSPECTION

The comments on Lead System Inspection relate to Appendix C in the DRAFT NUREG-1927, Revision 1. Note that in the FINAL NUREG-1927, Revision 1, the guidance on this topic is now incorporated in Chapter 3, "Aging Management Review."

11.1 Comments from Robert Einziger/NWTRB

Comment 11.1.1

Comment: P-CI, L32 - A couple of examples are given but specific guidance as to how these parameters should be taken into account when determining the lead system should be given.

Response: Upon consideration of the comments on lead system inspection, Appendix C was incorporated into new discussions regarding "Pre-Application Inspections" in Sections 3.4.1.2 and 3.6.1.10. The staff recognizes that the guidance in NUREG-1927 is generic. Therefore, in the revised Section 3.4.1.2, the staff has added clarification that applicants are encouraged to consider the criteria for the elements of an aging management programs (Section 3.6.1) when defining the scope, methods and acceptance criteria for pre-application inspections. The example aging management programs in Appendix B are identified as adequate examples for defining the scope, methods, and acceptance criteria for pre-application inspections of welded stainless steel canisters and reinforced concrete structures. Additional details are provided to clarify that the reviewer should ensure that the application provides a description of any initiated corrective actions (including results from actions to verify extent of condition) due to conditions identified in the pre-application inspection(s). The guidance further encourages applicants to discuss their considerations for selecting the system(s) to inspect with NRC staff in pre-application meetings before submitting the renewal application.

Comment 11.1.2

Comment: P-C2, L7 - The tendency will be to try and get all the necessary information by examining one system. The sentence starting "Different lead systems…" should be emphasized in bold.

Response: The staff recognizes that multiple systems may need to be inspected to adequately determine the extent of aging for different operable degradation modes. This consideration is reflected in the guidance. The use of emphasis may create confusion as to what sections in NUREG-1927 require most attention. The staff considers the entire guidance document to provide equally valuable information. The staff determined that changes to the guidance are not necessary.

Comment 11.1.3

Comment: P-C2, L 14-18 - This is sort of a chicken or the egg situation. Shouldn't the examination of the lead system be held to the same criteria as specified in the AMP for that type of inspection? On the other hand the AMP won't be available until the application is submitted. What is the value of the lead inspection if it isn't held to the same standards as the future

inspections? Suggest providing some guidance on the inspection techniques to be used for the lead system inspection.

Response: See the response to Comment 11.1.1.

Comment 11.1.4

Comment: P-C2 L-19 - Some explanation should be given to what needs to be provided by the applicant to address the issue of "when a visual examination appropriate.

Response: See the response to Comment 11.1.1.

Comment 11.1.5

Comment: P-C3, L15 - Isn't the use of "subset of sites considered bounding…" a contradiction to the guidance on page C2 that says "applicant should not refer to inspections done at other sites"? Please reconcile.

Response: See the response to Comment 11.1.1. The staff has provided clarification in Section 3.4.1.2 that, although CoC holders do not have the authority to conduct pre-application inspections at general licensee ISFSI sites, the CoC holder has the responsibility to provide the technical basis for the proposed approach to aging management at multiple sites. Therefore, pre-application inspections are likely only practical at a subset of sites within the CoC. In this case, the reviewer should ensure that the chosen subset of sites is bounding with respect to the susceptibility of the various potential aging effects. CoC holders could work within their user groups to identify bounding systems for the pre-application inspections.

To reconcile using a subset of sites for the review of a CoC renewal application, the guidance clarifies that the reviewer should ensure that aging management programs include baseline inspections at each site upon entering the period extended operation. During baseline inspections, or the first inspections conducted within the AMPs, each licensee should assess the condition of SSCs (1) to confirm the results of the pre-application inspections that were conducted at other sites are bounding, or (2) to verify the adequacy of the AMPs and the conclusions of the TLAAs when pre-application inspections were not performed. Also, the reviewer should consider whether uncertainties in SSC degradation due to the lack of a pre-application inspection warrant conditions in the license or CoC to require specific inspections immediately open entering the period of extended operation.

11.2 Comments from Nuclear Energy Institute

Comment 11.2.1

Comment: Throughout the guidance document, and especially in Appendix C the terminology "lead system inspection", "baseline inspection" and inspections associated with aging management programs are identified. The purpose of inspections associated with example aging management programs (AMPs) is clearly articulated with defined acceptance criteria contained in Appendix B. However, additional clarification on the purpose, scope and

acceptance criteria for the lead system inspections and the base line inspections (i.e., those inspection proposed to occur prior to the period of extended operation) is needed.

Response: See the response to Comment 11.1.1.

Comment 11.2.2

Comment: Section 3.6.1 and Appendix C contain recommendations for the performance of a lead system inspection as part of the specific license or Certificate of Compliance (CoC) renewal application. However, the Part 72 regulations have no requirement to perform such an inspection. Moreover, the CoC holder has no legal authority to compel any general licensee to perform such an inspection if that inspection is not already part of the licensing basis described in the storage system CoC or associated FSAR. Additionally, defining a "lead system" that would be bounding for all users of a single CoC, which may have several hundred loaded systems at multiple sites, with numerous unique geographical and environmental conditions, various materials of construction, and widely different in-service storage durations, is not workable.

Response: See the responses to Comments 11.1.1 and 11.1.5.

Comment 11.2.3

Comment: Appendix C uses the term "lead system inspection" interchangeably for both the pre-application inspection (page C-2, lines 24-27) and for the determination of the lead system for baseline inspections at each individual Independent Spent Fuel Storage Installation (ISFSI) (page C-1, Section C.2 and page C-1, lines 15-24) for the purposes of implementing the aging management program. The multiple uses of the term "lead system inspection" is confusing with respect to its meaning for CoC holders and licensees.

Response: See the responses to Comments 11.1.1 and 11.1.5.

Comment 11.2.4

Comment: When the NRC approved the change to 10CFR72 to allow for 40-year license and CoC terms (76 FR 8872) a commenter asked for the NRC to clarify when aging management requirements apply to casks. The NRC response was as follows:

> Aging management requirements only apply after [emphasis added] the cask is in service for the length of time equal to the term certified by the cask's initial CoC. For example, if the term of the initial CoC is 20 years, and a cask is placed into service at the end of the 19th year, then the general licensee would need to begin implementing the appropriate aging management requirements at the end of the 39th year, assuming the CoC was renewed.

Stipulating an inspection prior to the period of extended operation (PEO) should be assessed against the backfit provisions in 10 CFR 72.62(a). The current CoCs that have been approved for a period of twenty years have been certified to be safe without the need for additional inspection (other than those already specified in the storage system CoC or FSAR).

Response: The staff agrees that 10 CFR Part 72 regulations do not include a requirement to perform a pre-application inspection before the period of extended operation, therefore

assessment against the provisions of 10 CFR 72.62(a) is not necessary. As discussed in the response to Comment 11.1.1, the staff revised Section 3.6.1.10 to incorporate the previous Appendix C guidance on lead system inspection into new discussion on pre-application inspections. The staff does not characterize a pre-application inspection as a requirement, but as a means by which an applicant can demonstrate that an aging effect does or does not require management, and verify the condition of SSCs and SSC subcomponents are as expected. The pre-application inspection is performed before submittal of the specific-license or CoC renewal application, and the inspection results become part of the technical bases for renewal. The pre-application inspection provides valuable operating experience, which the NRC staff considers in conjunction with other relevant operating experience, it its review of the renewal application.

Comment 11.2.5

Comment: The NEI Dry Storage Task Force has given careful consideration to the following proposed terminology and approach for system inspections as a part of license renewal. The following terms, not currently included in the NUREG-1927 glossary, are defined below and will be included in NEI 14-03. The NRC may also consider adding these terms to the glossary in NUREG-1927.

Pre-application inspection: This is an inspection performed at the discretion of the licensee and/or CoC holder prior to submittal of the renewal application to (1) determine whether any SSCs have undergone unanticipated degradation, (2) to confirm existing TLAAs and inform the extension of those TLAAs throughout the period of extended operation (PEO), and (3) to help determine the appropriate AMPs needed through the PEO.

Baseline inspection: This is the first inspection of lead components in accordance with the AMP defined by the renewed site-specific license or CoC.

Lead-component (replacement for lead system): The lead component is that SSC or subcomponent of an SSC at an ISFSI that is determined to be susceptible to the aging mechanism identified and for which an AMP is applicable.

Response: See the responses to Comments 11.1.1 and 11.1.5. The ideas of a pre-application inspection and a baseline inspection have been clarified and included in Sections 3.4.1.2 and 3.6.1.10. The terminology of a “lead” system or SSC is no longer included in the guidance. However, the revised guidance in Section 3.4.1.2 and Section 3.6.1.4 (“Detection of Aging Effects”) on sample size has been expanded to include guidance on selecting SSCs for inspection based on parameters that may contribute to the potential aging mechanisms and effects for those SSCs. In addition, the staff added definitions for “pre-application inspection” and “baseline inspection” to the Glossary in Chapter 5.

Comment 11.2.6

Comment: With these definitions, the AMP identifies the inspection of in-scope SSCs and subcomponents through the PEO being requested. The licensee then implements the AMP by identifying the lead component(s) (for their ISFSI) to determine which system to inspect for a given SSC or subcomponent within the scope of that AMP. That inspection is performed in accordance with the inspection requirements and acceptance criteria defined in the AMP which has been approved by the NRC as part of the license renewal application. The licensee should utilize previous lead component inspections, possibly conducted at other sites to determine the

applicability of those inspections to their individual site for the purposes of satisfying the requirements of the AMP.

Response: See the response to Comment 11.2.5.

Comment 11.2.7

Comment: Additionally, the licensee should have the flexibility to optimize inspections by allowing a component that is similar in susceptibility to the lead component to be the component inspected based on the accessibility to the storage system and As Low As Reasonably Achievable (ALARA) considerations. This will allow for a more efficient use of personnel and resources, while eliminating unnecessary worker radiation exposure and cost.

Response: Although the licensee may have the flexibility to inspect different SSCs or SSC subcomponents during subsequent inspections (as it may not be restricted by a license or CoC condition or technical specification), the staff notes that the selection of different components for the inspections defined in the AMP relative to the components inspected in the pre-application inspection or the baseline inspection may present issues for purposes of monitoring and trending (considering the limited base of operating experience available). If subcomponents from different systems will be inspected throughout the period of extended operation, the licensee should ensure that a given operable degradation mode is being adequately trended (i.e., consistent with the licensee's approved quality assurance program). The staff agrees that consideration should be given to minimizing worker dose radiation exposure when selecting a subcomponent for inspection. As discussed in the response to Comment 11.2.5, the revised guidance in Section 3.4.1.2 and Section 3.6.1.4 has been expanded to include guidance on selecting SSCs for inspection. The staff also added clarification to the monitoring and trending guidance in Section 3.6.1.5.

Comment 11.2.8

Comment: Comment (Surrogate inspections to inform AMPs): Section C.4 of Appendix C stipulates that use of surrogates for lead system inspections cannot be referred to by an applicant. However, it is anticipated that as general and site-specific licensees enter the period of extended operation and additional inspections are conducted as part of an aging management program, a population of data will be created and shared as operating experience that will properly inform licensees with ISFSIs entering the PEO later of the rate of specific aging effects for their sites, based upon their site environmental conditions. Additionally, ongoing research and guidance is being developed, such as the EPRI Susceptibility Assessment and CISCC aging management program guidance, which will provide criteria to licensees to determine their relative susceptibility to CISCC compared to another site and other CISCC aging management insight. Therefore, it is requested that Section C.4 be revised to allow the use of surrogate inspections to inform the AMP. This concept is very important to the effective implementation of learning aging management. NEI 14-03, Revision 1 will propose an industry-wide operating experience sharing program in support of this approach.

Response: See the responses to Comments 11.2.5 and 11.2.7. With regard to use of surrogates, the revised guidance in Section 3.4.1.2 includes a caveat that the use of surrogates may only be accepted when the technical basis is supported by substantial operating experience. This language is meant to convey that (1) there is likely not an adequate base of operating experience at the current time to enable use of surrogate information, but (2) it is expected that an adequate base of operating experience will be established in the future, as

specific and general licensees enter the period of extended operation and conduct aging management inspections. The staff commends NEI's work on establishing an industry-wide database for sharing ISFSI aging-management operating experience between licensees.

Comment 11.2.9

Comment: C-1, 12. Add in "….SSC's in the renewal application" and within the licensee's corrective action program.

Response: See the response to Comment 11.1.1 for additional details on the revised discussion pertaining to corrective actions.

Comment 11.2.10

Comment: C-1, 28. Suggest adding: "with the exception of transfer casks or other similar SSCs which are leased or are otherwise not actually on site."

Response: See the response to Comment 11.1.1. Section 3.4.1.2 was revised to clarify that pre-application inspections may not include transfer casks or other similar SSCs which are leased or otherwise not actually on-site. The latter SSCs are generally subject to maintenance requirements before use. Records from these maintenance activities may be included in the application in support of their respective aging management programs.

Comment 11.2.11

Comment: C-1, 41. DSS should be "DSSs."

Response: The comment is no longer applicable because the guidance has been revised, as discussed in Comment 11.1.1.

Comment 11.2.12

Comment: C-2, 24-25. Add a reference to Subsection C.5 for further discussion on this issue for CoC renewals.

Response: The comment is no longer applicable because the guidance has been revised, as discussed in Comment 11.1.1.

Comment 11.2.13

Comment: C-2, 36-40. DSS with the same materials, fabrication practices, and design modifications, in use at different sites, are more common than this guidance implies. The guidance can emphasize that use of surrogates should be carefully considered and must clearly justify the bounding scenario, but use of surrogates should not be dismissed out of hand.

Response: See the response to Comment 11.2.8.

Comment 11.2.14

Comment: C-2, 36. While there is very limited operating experience specific to ISFSI DSS inspections, there is significant operating experience with the materials (stainless steels) and

environments (chloride in atmosphere) of concern. This experience along with additional research (CISCC Testing, environmental monitoring), and model results has been factored into draft Susceptibility Assessment Criteria that will be published by EPRI in September 2015. These criteria can point the industry to specific canisters where material and environmental conditions indicate higher likelihood of CISCC. The concerns expressed relative to materials, fabrication practices, and design modifications would also be relevant to the comparison of two different canisters at the same site. Differences in environmental conditions are indeed critical to CISCC susceptibility and thus the industry should have flexibility to devote more inspection resources to sites where conditions are more conducive to CISCC initiation and growth. This section should be modified to provide flexibility.

Response: The staff agrees that there is generic (i.e., non DSS- or ISFSI-specific) operating experience with the materials and environments of concern for some aging mechanisms, which NUREG-1927 states could be cited in support of proposed aging management programs (see Section 2.1 and Section 3.4.1.2). See also the responses to Comments 11.1.5 and 11.2.8 regarding the use of surrogate inspections.

Comment 11.2.15

Comment: C-2, 37. Should add a qualifier for allowing bounding sites in the future after industry gets more data.

Response: See the responses to Comments 11.1.5 and 11.2.8 regarding the use of bounding and surrogate inspections.

Comment 11.2.16

Comment: C-3, 10-11. CoC holders are not known to own any currently deployed DSSs, so the phrase “the CoC holder may not own the deployed DSSs at general-licensed ISFSI sites” really should be changed to “the CoC holder very rarely owns the deployed DSSs at general-licensed ISFSI sites” to more properly set the stage for the remainder of the paragraph.

Response: The comment is no longer applicable because the guidance has been revised, as discussed in Comment 11.1.1.

Comment 11.2.17

Comment: C-3, 12-13. The sentence, “This demonstration, and thus, the lead system inspection to support it, is nevertheless the responsibility of the CoC holder as applicant,” cannot be included. The guidance can encourage a collegial arrangement and emphasize the importance of lead inspections, as it very much does, but it cannot assign this responsibility on an entity which has no legal authority to meet it.

Response: See the responses to Comment 11.1.1 and Comment 11.1.5.

CHAPTER 12: RESPONSES TO PUBLIC COMMENTS ON USE OF A HIGH BURNUP FUEL DEMONSTRATION PROGRAM AS A SURVEILLANCE TOOL

The comments on Use of a High Burnup Fuel Demonstration Program as a Surveillance Tool relate to Appendix D to NUREG-1927, Revision 1.

12.1 Comments from Nuclear Energy Institute

Comment 12.1.1

Comment: XX, Appendix D. The discussion of fuel cladding ductility and hydride re-orientation in Appendix D is not consistent with respect to the latest research findings, such as information presented at the REGCON2014, the 2015 Used Fuel Management Conference and the mechanical fatigue testing being conducted at Oak Ridge National Lab.

Response: The guidance in Appendix D is equivalent to that presented in Interim Staff Guidance (ISG)-24, "The Use of a Demonstration Program as a Surveillance Tool for Confirmation of Integrity for Continued Storage of High-Burnup Fuel beyond 20 Years," which was issued for public comment and finalized in July 2014. The staff monitors new research on high-burnup fuel, including that sponsored by the NRC on "Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications" (NUREG/CR-7198, May 2015, ADAMS Accession No. ML15139A389). The research presented in NUREG/CR-7198 (work performed by Oak Ridge National Laboratory) does not discuss mechanical fatigue testing of high-burnup fuel subjected to radial-hydride reorientation, as that research is ongoing and results are expected to be published in 2016. However, the staff has noted research programs in Section 3.4.1.4 of the guidance, including the work being conducted at Oak Ridge National Laboratory. The comment is unclear as to what additional research presented at the cited conferences (REGCON 2014, 2015 Used Fuel Management) should be considered in revision to the publicly vetted Appendix D.

The staff clarified the references listed in Section 3.4.1.4 of the guidance.

Comment 12.1.2

Comment: D-1, 31. The statement that the fuel could be too brittle to retrieve is misleading. There are no design basis events under normal, off-normal or accident conditions that would apply enough load to fracture the cladding, even if it is brittle. Also, methods for retrieving fuel with severely damaged cladding do exist and could be deployed.

Response: The staff agrees that expected design-bases loads under normal and off-normal conditions of storage are not expected to compromise cladding integrity. However, the guidance in Appendix D, which is consistent with ISG-24, is clear that the concern for radial hydride reorientation is associated with the retrievability of HBU fuel on an assembly-basis. The applicant should evaluate the applicability of this concern based on the consideration of "ready-retrieval" as defined in the design bases. The staff has determined that changes to the guidance are not necessary.

The staff also agrees that methods for retrieving fuel with severely damaged cladding exist.

Comment 12.1.3

Comment: D-2, 39. Change “LWR” to “HBU.”

Response: The staff agrees with the comment and has made the suggested change.

CHAPTER 13: RESPONSES TO PUBLIC COMMENTS ON CONSIDERATIONS FOR RENEWALS OF CERTIFICATES OF COMPLIANCE

The comments on Considerations for Renewals of Certificates of Compliance relate to Appendix E to NUREG-1927, Revision 1.

13.1 Comments from Nuclear Energy Institute

Comment 13.1.1

Comment: Comment (Application of CoC AMPs to multiple sites): Renewal of CoCs and the relationship to the general licensee warrant special consideration that is unique from specific ISFSI license renewal. In the case of a specific license renewal, the licensee is the applicant. However, for renewal of CoCs, the applicant is the certificate holder, and the licensees are those Part 50 license-holders who have chosen to use the dry storage system covered by the individual CoC at their ISFSI. General licensees may have multiple systems, loaded over several decades, with individual casks entering the period of extended operation at different times, based upon when each individual cask was loaded. General licensees also may be using several different storage system designs certified under different CoCs at the same site ISFSI.

The example AMPs contained in Appendix B do not reflect the possibility that different general licensees' ISFSIs may experience distinctly different environmental conditions. Therefore AMPs approved generically under a renewed CoC could perhaps be implemented with different frequencies or applicability among general licensees with less aggressive environments than the limiting case environment considered by the CoC holder in developing the AMPs. The example AMPs should be updated with guidance on how individual general licensees should determine the applicability of AMPs to their specific site depending on their site-specific environment and the process for modifying the AMPs under 10 CFR 72.48 and documenting those AMP modifications in the site 72.212 Evaluation Report. For example, management of chloride-induced stress corrosion cracking (CISCC) of a stainless steel canister by a general licensee for an ISFSI in a fresh water environment that has minimal chloride bearing substances in the ISFSI area should not include implementing the associated AMP inspections of the canister as frequently as a licensee with canisters at an ISFSI site located near a salt-water environment.

Response: The staff disagrees with the comment that the example AMPs should be updated with guidance on how individual general licensees should determine the applicability of AMPs to their specific site depending on their site-specific environment. NRC staff recognizes the general license framework, where many cask users (general licensees), located in various environments may be using the same AMPs for the CoC. However, the example AMPs in Appendix B are not intended to provide criteria for how a user in a different or "less aggressive" environment for that particular aging mechanism/effect should modify the AMP or aging management activities. CoC applicants may propose as part of their AMPs, how the AMPs can be modified by users for certain environments, and the criteria for doing so. Appendix E already notes this in Section E.1. Appendix E also includes guidance on how users will update their 10 CFR 72.212, "Conditions of General License Issued under § 72.210," reports to document how they are meeting the terms, conditions, and specifications of the renewed CoC. Appendix E notes the possibility for the cask FSAR specifying the applicability of the AMP or certain AMP details (e.g., inspection frequency) to certain environmental conditions. The staff

has added an additional clarification to this section to note that the general licensees must evaluate any deviations to the cask FSAR under 10 CFR 72.48, “Changes, Tests, and Experiments.”

Comment 13.1.2

Comment: E-1, 35-36. This statement should be revised to recognize that general licensees that are in timely renewal will have a grace period after the effective date of the renewed CoC to update the 72.212 Report.

Response: The staff agrees with the comment. There is language in Section 3.6.3 that recognizes the situation where development of the infrastructure for AMP implementation and actual AMP implementation before the period of extended operation may not be possible, considering timely renewal provisions. The NRC staff added similar language to Appendix E, in terms of AMP implementation and the update of the 10 CFR 72.212(b)(5) report.

Comment 13.1.3

Comment: E-1, New. Suggest the NRC adding an expectation that AMP-related information in the cask FSAR will be implemented by general licensees as-written, unless modified under the provisions of 10 CFR 72.48.

Response: The NRC staff agrees with the comment and made the recommended change.

CHAPTER 14: RESPONSES TO PUBLIC COMMENTS ON STORAGE TERMS

The comments on Storage Terms relate to Appendix F to NUREG-1927, Revision 1.

14.1 Comments from Nuclear Energy Institute

Comment 14.1.1

Comment: XX, Appendix F. This appendix should be expanded to provide more examples such as were contained in the Statements of Consideration with the initial rulemaking to change the license term to 40 years (76 FR 8872) and the April 8, 2015, ACRS meeting.

Response: The staff disagrees with the comment. The general examples provided in F.2 adequately capture the various scenarios that were provided in the Statements of Consideration (SOC) for the 2011 rulemaking for 10 CFR Part 72 (76 FR 8872; February 16, 2011). The flowchart in Figure F-1 is a tool to help a user calculate the storage term for the various scenarios. The 2011 SOCs are already referenced in Appendix F, so a reader may refer to them to see the specific scenarios included in the SOC. No changes were made to the guidance.

Comment 14.1.2

Comment: F-1, 13-14. This is contrary to the second sentence of 72.212(a)(3), which reads, "For any cask placed into service during the final renewal term of a Certificate of Compliance, or during the term of a Certificate of Compliance that was not renewed, the general license for that cask shall terminate after a storage period not to exceed the length of the term certified by the cask's Certificate of Compliance."

Response: See the response to Comment 3.2.5.

Comment 14.1.3

Comment: F-2, Figure F-1. This chart says that all casks loaded during the initial license period of a license that is renewed have the same storage period, no matter when there were loaded during the initial license period. Is this the intent?

Response: The flowchart in Figure F-1 does not indicate what the comment suggests. Following the flowchart, answering "yes" for "loaded during initial CoC term?" and "yes" for "renewed?," leads to the box in the bottom left of the figure. "A" indicates the **remaining time in the initial license period** [emphasis added], based on when the system was loaded in the initial license period. So, the storage term would vary based on when exactly the system was loaded during the initial period. No changes were made to the figure.

Comment 14.1.4

Comment: F-2, Figure F-1. Rectangular box in lower left corner – the text "A = remaining time in the initial license period" change to "A = remaining time in the initial CoC term"

Response: NRC staff agrees with the comment and has made the recommended change.