COMMUNICATION STRATEGY FOR THE ENHANCEMENT OF PUBLIC AWARENESS REGARDING POWER REACTORS TRANSITIONING TO DECOMMISSIONING
February 2015

Introduction

The objective of this communication plan is to outline the U.S. Nuclear Regulatory Commission (NRC) strategy for communicating the key messages regarding the NRC process and practices for public and stakeholder engagement during the decommissioning of nuclear power plants, as well as to provide a resource for addressing frequently asked questions in this area.

Background

To date, the NRC has provided oversight for the decommissioning of 11 nuclear power plants. When a licensee decides to permanently cease operations at a nuclear power plant, the facility must be decommissioned by safely removing it from service and reducing residual radioactivity to a level that permits release of the property and termination of the operating license. The NRC has strict rules governing nuclear power plant decommissioning, involving cleanup of radioactively contaminated plant systems and structures, and removal of the radioactive fuel. These requirements protect workers and the public during the entire decommissioning process and the public after the license is terminated. With the recent increase in the number of power reactors beginning the decommissioning process and significant changes that have occurred in the regulations since 1996, there is an increased public interest in the decommissioning process.

Public involvement in the NRC's activities is a cornerstone of strong, transparent regulation of the nuclear industry. The NRC recognizes the public's interest in the proper regulation of nuclear activities and provides various opportunities for citizens to make their opinions known. The NRC seeks to elicit public involvement early in any regulatory process, including decommissioning, so that safety concerns that may affect a community can be resolved in a timely and practical manner. This process is considered vital to assuring the public that the NRC is making sound, balanced decisions about nuclear safety. Consistent with this policy, the NRC frequently hosts meetings with interested stakeholders, including members of the public, non-government organizations, and local and state government officials in order to discuss any topics that may be of interest.

The NRC recognizes the need and desire for community involvement in the decommissioning of a nuclear power plant. Decommissioning is a complex process and the NRC believes that decommissioning impacts need to be vetted within the local community. However, the NRC was created by the Congress to be an independent regulator charged with ensuring public health and safety and protecting the environment. As an independent regulator, the NRC ensures that all members of the public are given a fair and equal opportunity to comment on a licensee's decommissioning and license termination plans. Therefore, the NRC does not officially recognize or endorse any specific special interest group, public or private organization, community group, coalition, or individual. This approach assures that one or more organizations does not dominate the public forum and allows members of the public to provide alternative and differing viewpoints and comments to the NRC.

Current Public Outreach Tools for Power Reactor Decommissioning

Opportunities for the Public to Comment on Decommissioning Documents

The NRC regulations currently offer the public several opportunities to review and provide comments on licensee documents during the decommissioning process. Specifically, under the NRC regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.82, the NRC is required to publish a notice of the receipt of the licensee's Post-Shutdown Decommissioning Activities Report (PSDAR) and the License Termination Plan (LTP), make the PSDAR and LTP available for public comment, schedule separate meetings in the vicinity of the location of the licensed facility to discuss both the PSDAR and LTP within 60 days of receipt, and publish a notice of the meetings in the *Federal Register* and another forum readily accessible to individuals in the vicinity of the site. An example of this type of *Federal Register* (FR) notice for a PSDAR is the NRC's Notice of Public Meeting and Availability of Report published in the *Federal Register* on April 1, 2013, for the Kewaunee Power Station (78 FR 19540).

The PSDAR serves as the main planning tool for the decommissioning process, including the estimated cost of the decommissioning activities. By regulation, a licensee has two years to submit the PSDAR from the time operations are permanently ceased at the nuclear power plant (i.e., the beginning of decommissioning process). The licensee must submit its LTP at least two years before the expected license termination request (i.e., the end of the decommissioning process). In addition, because the NRC approves the LTP by amending the license, a hearing opportunity occurs for the LTP and follows the normal amendment process.

Community Outreach and Advisory Groups

For many years, the NRC has recommended that licensees involved in decommissioning activities form a community committee to obtain local citizen views and concerns regarding the decommissioning process and spent fuel storage issues. It has been the NRC view that those licensees who actively engage the community maintain better relations with the local citizens. NRC guidance related to creating a site-specific community advisory board can be found in NUREG-1757, "Consolidated Decommissioning Guidance," Appendix M, "Overview of the Restricted Use and Alternate Criteria Provisions of 10 CFR Part 20, Subpart E," Section M.6. However, it should be noted that Appendix M applies *only* to sites requesting to terminate a license with restricted use conditions, which has not occurred. To date, all nuclear power plant facilities have been released for unrestricted use at the conclusion of the decommissioning activities. In addition, Appendix M does not *require* licensees to create a community advisory board, but only provides recommendations for methods of soliciting public advice. Nonetheless, Section M.6 contains useful guidance and suggestions for effective public involvement in the decommissioning process that could be adopted by any licensee.

Experience gained and lessons learned from prior decommissioning projects have been well documented by the nuclear industry. In 2005, the Electric Power Research Institute (EPRI) published the "Maine Yankee Decommissioning – Experience Report – Detailed Experience 1997 – 2004" (EPRI 1011734). In this lessons learned report, industry recognized that engaging the local community and officially forming a Community Advisory Panel or Board (CAP/CAB) is a good practice. Specifically, the EPRI report states that "the Maine Yankee Community Advisory Panel was established in 1997 to enhance opportunities for public involvement in the decommissioning process of Maine Yankee. The CAP represents the local community. By thoroughly reviewing the decommissioning process, the CAP is in a position to advise Maine Yankee on key issues of concern to the local community."

In addition, the NRC held a decommissioning status meeting in October 2005, during which members of the Maine Yankee CAP were invited to speak to the Commission regarding their experiences. At this meeting, the Vice Chairman of the Maine Yankee CAP concluded that CAPs "provide an important window for the public in the process of decommissioning, and provide the opportunity for issues of local concern to be addressed both within and without the strict process defined by the regulations. As a result, in our decommissioning, a level of trust was gained that had evaded Maine Yankee for the previous 24 years of operation." The complete information presented during that meeting can be found at http://www.nrc.gov/reading-rm/doc-collections/commission/slides/2005/20051018/hudson-material-decommissioning.pdf.

Since the decommissioning of Maine Yankee, licensees have employed a CAP or CAB at many other sites, including Connecticut Yankee, Yankee Rowe, Big Rock Point, Millstone, Humboldt Bay, and others. For all of the nuclear plants currently entering into the decommissioning process, the NRC has strongly encouraged the licensees to establish a CAP/CAB for the decommissioning effort in order to enhance communications with the local communities and stakeholders. In addition, as part of our role as an independent regulator, the NRC frequently attends CAP/CAB meetings to address questions or concerns from the community members.

Decommissioning Lessons Learned and Additional Guidance Documents

The NRC routinely documents experience gained and lessons learned. Generally, lessons learned are any items that could be of interest and benefit to many licensees. Lessons learned include positive or negative discussions that are worth sharing with NRC licensees and stakeholders. For example, lessons learned can include significant and recurring NRC comments on licensing documents, issues that have been formally documented and evaluated by the staff, case studies providing useful site-specific examples or best practices for some aspects of decommissioning, or unsuccessful experiences that should be shared to avoid recurrence. The NRC staff has historically collected lessons learned during the decommissioning process and attempted to incorporate these insights into ongoing activities.

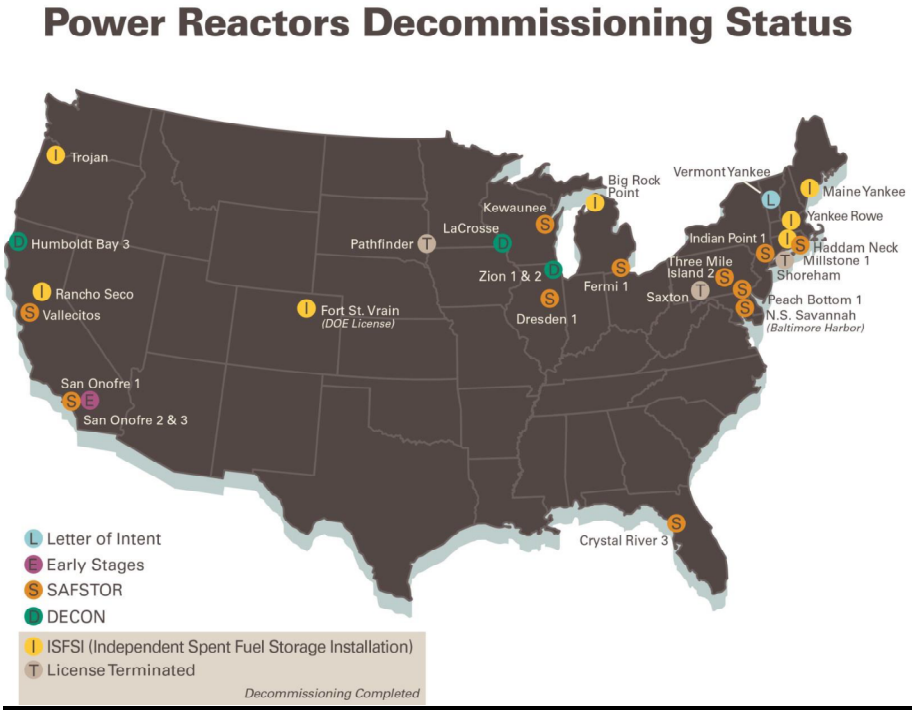
Additional References on Decommissioning Lessons Learned can be accessed on the NRC website. This is a preliminary collection of documents that the NRC, Fuel Cycle Facilities Forum, Nuclear Energy Institute, Electric Power Research Institute, and Organization of Agreement States have compiled to inform existing and future NRC licensees, NRC staff, States, and other interested stakeholders about potential lessons learned from past decommissioning actions. This list includes information such as: a) title of the document, b) list of author(s), c) summary of the content of each document, and d) details on the availability of each document in this bibliography. In addition, reports from the Decommissioning Lessons Learned Database can be accessed for the years 2002, 2004, 2005, 2006, and 2007.

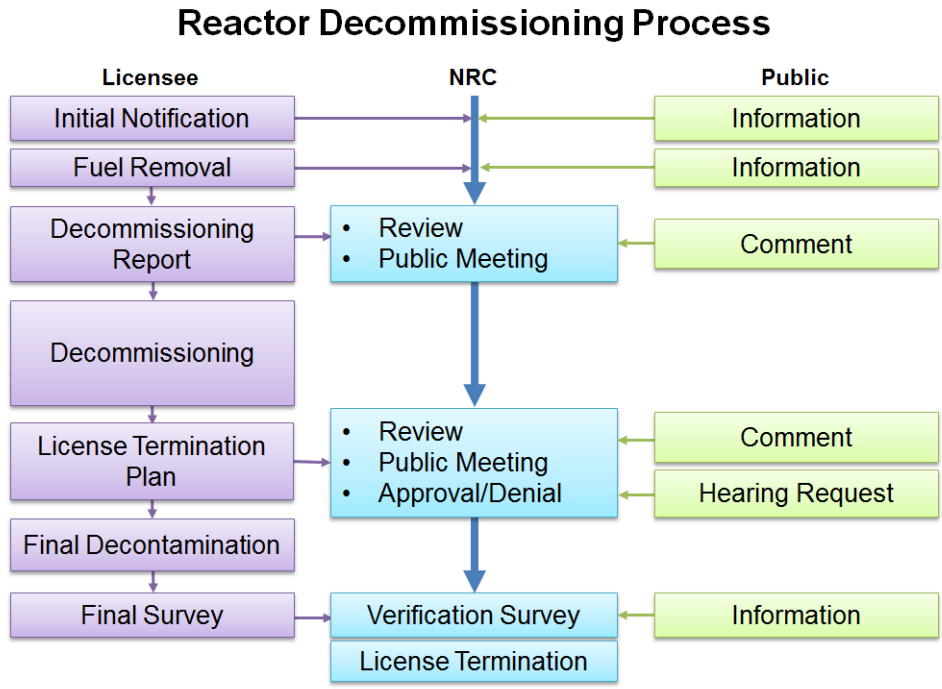
The NRC believes that adequate processes and the associated guidance documents are already in place to establish best practices regarding public and stakeholder engagement in the decommissioning of nuclear power plants within the bounds of the NRC's role as an independent regulator. However, many of these guidance documents are being updated to reflect additional lessons learned in decommissioning and will continue to be revised as new insights are gained. In addition, the existing practices within the agency promote community involvement in the decommissioning process to the extent practicable and attempt to capture lessons learned during each decommissioning activity to improve the overall regulatory process. As such, the NRC does not currently plan to create a Regulatory Issue Summary or any other type of additional guidance document to further discuss public participation in the

decommissioning process, but will continue to implement the process and procedures already established for the decommissioning of nuclear power plants.

Infographics Related to Decommissioning

Numerous graphics related to decommissioning are available; two of the more useful examples are included below.





In addition, the Decommissioning Program Brochure and the Decommissioning Backgrounder, which are released by the NRC's Office of Public Affairs and have been recently updated, give a high level overview of the decommissioning process and discuss involvement opportunities.

Frequently Asked Questions on Reactor Decommissioning

Because of the breadth and scope of questions regarding the decommissioning process, the information has been grouped into various topics that can be accessed by clicking on the appropriate link below to go to the associated appendix:

Note that several of the questions refer to specific decommissioning reactors; however, the information contained in the answers is generally applicable to all decommissioning facilities.

Please use these links, as well as those provided throughout the document to other external information sources to promote consistent communication on these topics.

Power Reactor Decommissioning Process

NUREG-1628 – This report contains a definition of decommissioning and a discussion of alternatives. It also provides a focus on decommissioning experiences in the United States and how the NRC regulates the decommissioning process. Questions related to spent fuel, low-level waste, and transportation related to decommissioning are answered. Questions related to license termination, the ultimate disposition of the facility, and finances for completing decommissioning and hazards associated with decommissioning are also addressed. This document provides responses to questions related to public involvement in decommissioning as well as providing the public with sources of obtaining additional information.

1. What are the unrestricted and restricted release criteria for power reactor decommissioning?

A: For Unrestricted Release

- Total Effective Dose Equivalent (TEDE) ≤ 25 mrem/yr (0.25 mSv/a)
- As Low As is Reasonably Achievable (ALARA)
- Average member of the critical group
- All pathways
- Period of performance = 1000 years

For Restricted Release

- Total Effective Dose Equivalent (TEDE) ≤ 25 mrem/yr (0.25 mSv/a)
- ALARA, with institutional controls in effect
- Legally enforceable institutional controls
- If institutional controls fail, doses do not exceed 100 mrem/yr (1 mSv/a), or 500 mrem/yr (5 mSv/a), under specific circumstances
- Financial assurance via an independent third party
- Licensee and NRC public input/outreach requirements

2. Will the NRC allow the resale of non-radioactive equipment and secondary side components (e.g., turbines, moisture separator reheaters, heat exchangers, condensers, intake pumps, intake piping, outfall piping, all associated piping and electrical components, etc.) since some of these are worth substantial amounts? Will the proceeds go to offset the cost of decommissioning?

A: The NRC regulates the safe use of radioactive materials and does not regulate commerce. As such, the NRC will ensure that the materials released from the site for unrestricted use meet radiological release requirements. If a licensee has non-radioactive assets, including plant components, equipment, and recyclable materials that can generate revenue, the use of this revenue is outside of the NRC's authority and is under the oversight of the state public utility commission. The licensee may also transfer or sell contaminated parts and equipment to other licensees for use in their nuclear facilities, within the limitations of the applicable transportation requirements.

3. Will there be public announcements when any "allowable" waste is to be released into the environment? What are the upper limits for releasing radiation and

chemicals into the environment during the decommissioning process? When were those limits established and what would trigger a process to reevaluate those limits?

A: During decommissioning, both liquid and airborne radiological releases will be monitored and are required to be maintained below the same radiological limits as when the plant was in operation. The radiological effluent release criteria were established in 10 CFR Part 20 many decades ago. The licensee will continue to provide the NRC with environmental effluent reports and the NRC will conduct inspections of this area throughout the decommissioning process. The results of the NRC inspections and any associated findings will be published in inspection reports that are publicly available. The NRC does not monitor or regulate the release of toxins or other forms of non-radiological waste as that is within the purview of the Environmental Protection Agency and other similar federal and state government organizations.

4. What level of on-site staffing will NRC provide during the decommissioning process, and in which areas of technical expertise and oversight?

A: Consistent with agency procedures, the NRC typically maintains a full time resident inspector onsite during part of the first year after permanent shutdown. The resident inspector oversees the plant transition from operation to permanent shutdown, in order to verify that the licensee complies with their license, technical specifications, and procedures. Generally, early in the first year after permanent shutdown, the Inspection Program is transferred from the Reactor Oversight Program to Inspection Manual Chapter (IMC) 2561, "Reactor Decommissioning." The contents of this IMC are publicly available and outline the oversight activities that the NRC staff will be involved in throughout decommissioning.

During the first year, the licensee prepares the plant for safe decommissioning. The actions taken by the licensee include the modification of systems, shipment of radioactive waste, emptying of tanks, draining of systems, and electrical isolation of components. As during plant operations, the resident inspection staff is supplemented with special inspection expertise as needed, which includes security, emergency response, health physics, environmental monitoring, and engineering. NRC inspections continue throughout decommissioning until the licensee demonstrates that the site meets the license termination requirements. The level of decommissioning inspections will be commensurate with the licensee's planned decommissioning activities.

5. Why doesn't the NRC review and approve the Post Shutdown Decommissioning Activities Report (PSDAR)?

The NRC does not approve the PSDAR. In Commission Paper, SECY 96-086, the Commission determined that decommissioning activities could be safely conducted under the current license conditions and restrictions. The Commission determined that a detailed decommissioning plan requiring NRC review and approval would be redundant to the activities already authorized by the Commission in the facility license. Any actions outside the license would require the licensee to request a license amendment and would ask them to appropriately justify the why the change was safe.

6. Will the NRC commence a rulemaking on decommissioning in order to add efficiency to the current process of requesting exemptions from operating plant requirements?

A: The NRC staff added to its Common Prioritization of Rulemaking a line item for a rulemaking on power reactor decommissioning transition. The Common Prioritization of Rulemaking is a budgeting tool that the NRC uses to assign resources to rulemaking activities, in light of the NRC's priorities and resource limitations. Separately, the NRC staff is in the process of determining the appropriate timeframe for the decommissioning transition rulemaking.

Even if this rulemaking were initiated immediately and implemented following the NRC rulemaking process of developing a regulatory basis; issuing a proposed rule; requesting, receiving, and addressing public comments on the proposed rule; and issuing a final rule; any improvements it may provide would have little impact on the plants that are currently transitioning to decommissioning. It would be available primarily for plants that may be considering permanently shutting down in the future. This consideration will be a factor in the NRC's determination of how to schedule such a rulemaking effort. The Commission recently directed the staff to report to the Commission its views on the need for an integrated rulemaking for decommissioning, based on lessons learned from the most recent operating closures.

7. What guidance or other regulatory documents does the NRC currently use to evaluate power reactors as they transition into decommissioning?

A: The main NRC guidance documents for decommissioning are:

NUREG-1496	Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities
NUREG-1700	Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans
NUREG-1727	NMSS Decommissioning Standard Review Plan
NUREG-1757	Consolidated Decommissioning Guidance
Regulatory Guide (RG) 1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors
RG 1.184	Decommissioning of Nuclear Power Reactors
RG 1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report
RG 4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning
RG 4.22	Decommissioning Planning During Operations

Decommissioning Funding

1. What is the minimum decommissioning funding assurance amount required by NRC? How is this amount calculated?

A: Licensees must estimate the minimum funding assurance amount needed for radiological decommissioning by using formulas in NRC regulations. As an alternative, licensees may also use a site-specific cost estimate to determine the minimum funding assurance amount needed, provided that this amount is greater than the amount derived from the NRC formulas. The minimum decommissioning funding assurance amount must be adjusted annually by applying the latest escalation factors for labor, energy and waste burial costs.

2. What are NRC decommissioning trust funds used for? Will there be sufficient funds to bring the site back to greenfield status?

A: NRC decommissioning trust funds are used for decommissioning as defined and regulated by the NRC. The NRC formulas address only those decommissioning costs needed to remove a facility or site safely from service and reduce radioactivity to safe levels to allow for termination of the license. However, the costs of removal of non-radiological systems and structures are not included in the NRC decommissioning cost formulas. In addition, the costs of managing and storing spent fuel on site until transfer to the Department of Energy for permanent disposal are not included in NRC decommissioning cost formulas. The NRC does not ensure that there are sufficient funds to bring a site to greenfield status.

3. What will be the NRC's response if a licensee's present decommissioning fund balance does not appear to be adequate?

A: At any time the NRC determines that unacceptable levels of decommissioning funding assurance exists, the NRC will determine a remedial course of action for any particular licensee on a case-by-case basis.

4. What types of actions might the NRC implement to ensure that licensees make any necessary adjustments to their financial assurance?

A: The NRC may require the licensee to take one or more actions, including but not limited to:

- Produce a funding guarantee from a parent company
- Develop a specific catch-up funding plan
- Provide more frequent or additional funding reports
- Make additional payments towards principal
- Obtain a surety bond

5. Does the NRC impose investment restrictions for decommissioning trust funds?

A: At the end of 2003, new investment regulations applicable to non-electric utility licensees became effective. The new restrictions prohibit decommissioning trust fund investments in licensee and nuclear sector securities or obligations. Absent State or Federal law restrictions applicable to decommissioning trusts, the NRC regulations require that investments be made in accordance with the "prudent investor standard."

6. Does the NRC regulate trustees for nuclear decommissioning trusts?

A: Three federal bank regulatory agencies - the Federal Deposit Insurance Corporation, the Office of the Comptroller of the Currency, and the Federal Reserve Board (Banking Agencies) - have oversight of the trust departments of financial institutions that typically act as trustees for nuclear decommissioning trusts. The Banking Agencies' examination process evaluates a trustee organization's compliance with its fiduciary obligations, including complying with applicable investment restrictions, whether they are the NRC's or those of any other agency or those of a state.

7. What happens if a licensee goes bankrupt?

A: If the licensee of a nuclear power plant or its parent company declares bankruptcy, it must still fulfill its decommissioning obligations. The NRC takes an active role to monitor and possibly intervene in bankruptcy proceedings through the Department of Justice to assure protection of public health and safety.

8. Are limited liability companies shielded in any way from decommissioning obligations?

A: Licensees, whether they are corporations or limited liability companies, are fully responsible for fulfilling their decommissioning obligations. Normally, their parent companies or stockholders are not licensed entities and therefore are not responsible, absent certain special circumstances that might render them liable.

Spent Fuel Considerations

High Burnup Spent Fuel

Nuclear fuel rods are ceramic pellets of uranium oxide (UO_2), about the size of a finger joint, stacked and sealed inside a long metal tube (cladding) about as big around as a Sharpie pen. The space between the pellets and cladding is filled with helium.

"Spent fuel" refers to fuel used in a commercial nuclear reactor that has been removed because it can no longer economically sustain a nuclear reaction. Burnup refers to the uranium consumed in the nuclear reaction. It is expressed in gigawatt-days per metric ton of uranium (GWd/MTU)—a measure of how long a fuel rod is in the core and the power level it reaches. "High burnup fuel" is in the reactor core for longer than "low burnup fuel."

In a reactor core, nuclear fuel undergoes physical and chemical changes. The UO_2 fissions—splits apart and releases energy—producing fission gas and fission products. The pellets swell, crack and release a small amount of fission gas inside the rod. The cladding also reacts with the reactor cooling water. This reaction forms an oxide layer on the outside (similar to rust) and produces hydrogen. About 15% of the hydrogen enters into the cladding metal. These processes occur slowly at first, then start to accelerate at about the time the fuel reaches burnup of 45 GWd/MTU. At this burnup level, the fuel is about 75% through its currently useful life.

The maximum burnup that NRC has allowed increased as technology advanced our ability to understand the changes the fuel undergoes in the reactor core. Our understanding of those changes is key to our ability to make safety decisions. Average fuel burn-ups have increased from around 35 GWd/MTU two decades ago, to over 45 GWd/MTU today. Anything over 45 GWd/MTU is considered high burnup.

Available information indicates that both low and high burnup spent fuel can be stored and transported safely. This information comes from operational experience with storage systems and short term tests. Considerable data is available on the properties of low burnup spent fuel and more confirmatory data is being obtained daily on high burnup fuel. That data is instrumental in enabling the NRC to make licensing decisions to allow spent fuel storage in specific dry cask designs.

As utilities began discharging high burnup spent fuel, dry storage cask designers amended their designs to account for physical differences in the spent fuel. The NRC approved several spent fuel storage amendments based on research data sufficient to show that high burnup fuel could be stored safely. Additional high burnup storage amendments are under review. The NRC fully expects data from current testing will become available to confirm that high burnup spent fuel will behave as expected and remain in a safe condition even in a transportation accident.

From the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, PSDAR Meeting

1. **High burnup fuel has been used at San Onofre since 1996. Was notice given to the public? Were workers made aware that high burnup fuel is more radioactive?**

   Answer

   New fuel designs, including high burnup fuel (HBF), undergo an NRC technical review, and are generally approved for use via a Topical Report that provides the technical evaluation of

the new fuel design and lists any limitations for its use. Once the new fuel design is approved with an associated Topical Report, all NRC reactor licensees are permitted to use that fuel design within their reactor core without requesting specific NRC approval, as long as the core continues to meet all design and safety limits. In an operating nuclear reactor, burnup is one of the many parameters that are considered in designing the fuel and core for each operating cycle. Many parameters are evaluated throughout the operating cycle to verify that design specific limits are met. Data-based, predictive tools are used to evaluate these parameters over the cycle. Throughout the cycle, physics testing is also done to confirm key physics parameters are consistent with predictions. When a new fuel is designed its use is limited by the data available to support the associated predictive tools. As such, burnup is limited for a particular fuel by the supporting predictive tools, the data supporting the predictive tools, and the requirement to not exceed any design limit. In an operating reactor, the main distinction between HBF and low burnup fuel (LBF) is the amount of exposure to which the fuel is subjected. The fuel is required to meet all safety limits at all times during the operating cycle.

a. High burnup fuel is hotter and "between 2 and 158 times more radioactive," requiring the waste to be cooled on-site in spent fuel pools for at least 12-15 years (rather than 5 years). Does the NRC agree with this statement? If not, how much more radioactive would the NRC say high burnup is?

Answer

HBF is typically defined as fuel with a burnup (a measure of the time a fuel assembly stays in the reactor core) greater than 45,000 megawatt-days per metric ton of initial uranium. LBF means any fuel with a lower exposure than this value. Average fuel burnups have increased from around 35,000 megawatt days per metric ton uranium two decades ago, to over 45,000 megawatt days per metric ton uranium today. Higher burnup fuel is thermally hotter and more radioactive than lower burnup fuel for a given cooling time. The difference in decay heat (a function of the fuel transferring heat to decrease its temperature over time) and radioactive source term depends on the difference in the fuel burnup (i.e., how long the fuel was being used in the reactor), the initial enrichment of the fuel, and the irradiation environment that the fuel was exposed to in the reactor. Moving from 35,000 megawatt days per metric ton uranium to 45,000 and 55,000 increases the overall heat source term or level by 35% and 78%, respectively, and increases the radiation source term or level by 33% and 72%, respectively.

HBF must be cooled longer than LBF before it can be placed into a dry storage system. How much longer depends on the difference in burnup, the specific dry storage system design, and the decay heat loading pattern of the fuel being used. As an example, for a 5.0 weight percent enriched (in Uranium-235) fuel assembly in one particular storage system, the required cooling time goes from 4.5, to 7, to 12 years, for a fuel burnup of 35,000, 45,000, and 55,000 megawatt days per metric ton uranium, respectively.

b. How does the presence of high burnup fuel affect the decommissioning process at San Onofre? What specific problems does this higher radioactive fuel present for waste storage in fuel pools and dry cask storage at San Onofre and how much longer will this radiation last? How will decommissioning be impacted by the current onsite storage of the spent fuel?

Answer:

HBF is decommissioned in accordance with the same regulatory requirements as other fuel types. The higher heat load of HBF will require the water circulation pumps in the spent fuel pool to circulate more water in order to efficiently reject the added heat from the HBF and maintain water temperatures. Once the HBF spent fuel is sufficiently cooled in the pool, it may be transferred from the fuel pool to the dry storage canisters in the Interim Spent Fuel Storage Installation (ISFSI), after which the spent fuel pool can be decommissioned consistent with the licensee's decommissioning strategy and schedule.

A dry cask storage system that is designed to handle HBF has been reviewed and approved for use at SONGS. This review included determining if the dry cask storage system provides adequate margins to safely store HBF. SCE is only able to store fuel within the fuel specifications and other limits in the associated certificate of compliance, which provides reasonable assurance of the safety of the stored fuel. The impact on the overall decommissioning schedule for SONGS will be dependent on the spent fuel management plan that the licensee will determine. To date, SCE has not provided their decommissioning and spent fuel management plans for the NRC to evaluate.

c. We understand the NRC staff is worried about short and long-term waste storage in dry casks of high burnup fuel and has initiated a new study to determine if it can safely be stored in dry casks. Is this report complete? Will it be released to the public, and when?

Answer

It is not clear what NRC study the question is addressing. Currently HBF is licensed to be stored in approved dry cask storage systems for an initial period of 20 years, with a potential extension of one or more 40 year intervals. However, with the delay in availability of a final repository for spent nuclear fuel it was determined that the dry storage of spent fuel might have to account for a considerably longer period of time than originally planned. As such, the NRC staff is examining the regulatory framework and potential technical issues related to extended dry storage and subsequent transportation of spent nuclear fuel for periods beyond the initial licensing and single renewal period (i.e., beyond ~60 years of storage). This analysis has been undertaken in order to identify potential changes needed to the associated regulations or guidance, in accordance with direction from the Commission in Staff Requirements Memorandum COMSECY-10-0007, "Project Plan for the Regulatory Program Review to Support Extended Storage and Transportation (EST) of Spent Nuclear Fuel" (see Agencywide Documents Access and Management System (ADAMS) Accession No. ML103400287).

As a first step in the EST Program, the NRC identified the technical information needs associated with extended dry fuel storage systems (see draft issued for public comment, at ADAMS Accession No. ML120580143) in order to understand potential degradation mechanisms, the level of knowledge about these mechanisms, and how that degradation would affect the ability of a dry storage component to fulfill its regulatory or safety function. The EST Program is expected to be complete in the next five to seven years, subject to any funding limitations that may occur. The report, once finalized, will be made available to the public.

d. One of the NRC concerns is that there is no way to monitor what is occurring inside the dry casks. How does the NRC propose to monitor the highly radioactive material inside of the dry casks? How many casks will be required to safely store all the high burnup fuel that is on site in both the spent fuel pools and dry casks at San Onofre? How much high burnup fuel is on site in fuel pools and dry cask at San Onofre?

Answer

The NRC collects data on the total amount of spent fuel stored at commercial facilities, like SONGS, throughout the country. This information, and much more concerning spent nuclear fuel, is available on the NRC website at http://www.nrc.gov/waste/spent-fuel-storage.html. The NRC does not independently maintain records showing how many fuel assemblies have been loaded into each cask that were specifically HBF. However, licensees and certificate holders are required to register each cask with the NRC pursuant to 10 CFR 72.212(b)(2). The cask registrations and information contained therein are subject to routine inspection by the NRC. When the last routine ISFSI inspection report was issued for SONGS on May 20, 2011, SCE had loaded 11 canisters that contained at least one HBF assembly out of the 55 canisters loaded on the ISFSI pad at SONGS (see ADAMS Accession No. ML111430612). For any given facility, more specific spent fuel information is considered security-sensitive and is therefore not disclosed to the public.

The NRC is actively monitoring the efforts of industry and the Department of Energy (DOE) to better understand fuel aging mechanisms and promote methods for monitoring the behavior of fuel inside a sealed dry cask. DOE is taking an active role in funding Nuclear Energy University Program (NEUP) projects to study this issue. In addition, DOE is sponsoring a demonstration test with a variety of HBFs to benchmark models of fuel behavior and obtain, through monitoring of the mockup test casks, the temperature and gases evolved which will indicate how the fuel is behaving. This demonstration will provide data that will allow the creation of enhanced models to predict the behavior of various types of spent fuel.

In terms of the number of dry casks that will be necessary to store the spent fuel at SONGS, it will depend on which dry cask storage system is used for the decommissioning effort and when the spent fuel is actually moved from the fuel pool to dry cask storage due to the different sizes of casks available and the actual heat being given off by the assemblies the licensee wants to load into the dry casks. This information is not yet available from SCE, but should be provided as a part of the SONGS spent fuel management plan.

e. We know that MOX fuel was used in San Onofre, Unit 1 and was removed to the GE Morris facility in Illinois. How and when was that done and under what permit was that done? If MOX fuel was transported away, can other high burnup fuel be moved from the site in the same way to the same or similar places?

Answer

Between March 1972 and September 1980, 270 fuel assemblies were shipped from SONGS to the GE Morris facility. Mixed oxide (MOX) fuel was not included as a part of these shipments. Specifically, Appendix A of the Technical Specifications for GE Morris

(see ADAMS Accession No. ML042180413) identifies that 270 stainless steel clad fuel assemblies are allowed to be stored at the GE Morris facility. Page 12 of a 2004 NRC Inspection Report (see ADAMS Accession No. ML040070255) indicates that the MOX fuel irradiated at SONGS is zircaloy-clad. Consequently, GE Morris is not allowed to store the MOX fuel from SONGS. The NRC report also indicates that the MOX fuel was stored in the SONGS Unit 1 spent fuel pool, and Page 9 of the 2004 NRC Inspection Report identifies that irradiated MOX fuel can be loaded into the Advanced NUHOMS dry cask storage system employed by SONGS. In addition, a 2011 NRC Inspection Report (see ADAMS Accession No. ML111430612) identified that eighteen of the dry storage canisters located on the SONGS ISFSI pad contain either irradiated fuel or greater than Class C radioactive waste from SONGS Unit 1.

SONGS can transfer spent fuel only to licensees/licensed facilities that are authorized to receive the spent fuel and have an agreement in place with SCE to accept the fuel. Information on SCE's plans for the spent fuel is not yet available, but should be provided as a part of the SONGS spent fuel management plan.

2. **The NRC has not approved the transport of dry storage casks nor even short-term dry cask storage (beyond 20 years) for high burnup fuel. Will the NRC continue to allow high burnup fuel use even though they do NOT have an approved safe solution to store or transport this waste - even short-term? High burnup fuel is more difficult to store and transport. In addition, there is no transport cask design approved to store high burnup fuel. The NRC currently licenses dry cask storage for high burnup for only 20 years. The current expiration date for SONGS is February 5, 2023.**

Answer

While HBF does possess a higher initial heat load that LBF, it can be safely stored and transported if the potential fuel degradation mechanisms are addressed and the NRC regulatory requirements are satisfied. The assertion that the NRC has not licensed any casks for transport of HBF or short term storage of HBF is incorrect. HBF is in dry cask storage at a number of reactor sites across the country. In addition, there are a limited number of transportation packages certified for the transport of spent nuclear fuel. Of those, some have HBF included in the authorized contents. For example, the Model No. UMS Universal Transport Cask package and the Model No. Hi-Star 180 transportation package have current transportation certificates of compliance which include HBF contents.

The NRC will continue to license the storage of HBF as long as the applicants can show that it can be done safely by meeting the applicable regulatory requirements. Currently HBF is licensed to be stored in approved dry cask storage systems for an initial period of 20 years, with a potential extension of one or more 40 year intervals. The NRC is developing a path forward for licensing the storage of HBF up to 60 years and beyond based on time limiting aging analysis, aging management plans, analysis of consequences of a variety of potential fuel behaviors, and test data provided by the applicants.

There are three types of dry cask storage systems for spent nuclear fuel: 1) storage only which have not been approved for transportation, 2) dual purpose systems that are designed for both storage and transport (most of these have been approved for storage of HBF only but some have also been approved for transport of HBF), and 3) canisterized systems where the canister may or may not be put into a new overpack for transport. The current dry storage cask design in use at SONGS is the Transnuclear (TN) NUHOMS

system, Certificate of Compliance 072-1029, Amendment 1, with a 24PT4-DSC canister. The Technical Specifications for this canister allow fuel burnup of up to 60,000 megaawatt days per metric ton uranium. Note that higher burnup fuel requires a longer required cooling time (years after discharge) in the spent fuel pool before the heat load has decreased enough to meet the Certificate's Technical Specifications for using the dry cask storage system.

The Coalition cites the Regulatory Information Conference (RIC) talk by a senior NRC staff member on the storage and transportation of HBF. Specifically, the statement on Slide 7 of the Argonne National Laboratory (ANL) results that says there is "enough data to determine there is a regulatory issue. Insufficient data to support a licensing position." In late 2012 the NRC completed an experimental program at ANL to determine at what temperature cladding goes from ductile (able to bend) to brittle condition. That program found that different types of cladding become brittle at different temperatures. Dry storage cask designers will therefore need to address the specific fuel cladding type and its temperature to show that HBF can stay intact during a transport accident.

3. Is removing the spent fuel pool considered part of the decommissioning process? If so, how can dry casks be transported without use of a spent fuel pool in cases where that may be needed? What vulnerabilities are there in San Onofre's spent fuel pools? What improvements could be made to improve safety? Will any of them be made? If so, when? If not, why not?

Answer

There are two spent fuel pools at SONGS, both of which were built during construction of the plants. Both of the spent fuel pools meet all regulatory requirements for the storage of new and spent nuclear fuel. Decontamination of the spent fuel pools, which may involve removal and disposal, is a decommissioning activity. Before starting to decommission systems and components needed for moving, unloading, and shipping the spent fuel (like the spent fuel pool), the licensee must develop a plan for removal of the spent fuel from the reactor site, and a plan for how the spent fuel will be managed until the time that DOE takes title to, and possession of, the spent fuel (these requirements are in 10 CFR 72.218). This plan is part of the licensee's program for managing and providing funding for management of spent fuel following permanent shutdown of the reactor until the time that DOE takes the fuel. The licensee is required (in 10 CFR 50.54(bb)) to submit this spent fuel management program to the NRC for review within 2 years following permanent shutdown.

The licensee's program for spent fuel management and its plan for removal of the spent fuel from the site would need to consider what equipment, systems, or facilities would be needed to place the spent fuel into an approved transportation package, before decommissioning these systems. For example, if the storage system is a canister-based system (the spent fuel is confined in a welded canister that is placed in a storage overpack that provides radiation shielding), the canister may be placed into a transportation overpack / package without the need for a spent fuel pool. If the storage system was not transportable, a spent fuel pool or dry transfer facility would be needed to repackage the spent fuel into an appropriate transportation package when necessary.

SONGS currently uses a dry canister based storage system for its spent fuel in conjunction with the spent fuel pools. Regarding transportability of this system, the 24PT1 canister is certified for transport (under CoC 9255; MP187 transportation cask), as is the 24PT4

canister (under CoC 9302; MP197 transportation cask). As discussed earlier, the NRC is currently reviewing an application for amendment to CoC 9302 (MP197 transportation cask) to allow transportation of the 32PTH2 canister.

4. **Is the utility allowed to decommission the reactor and support facilities, including the spent fuel pool, while spent fuel is still being stored at the ISFSI? If so, what happens if a spent fuel cask has a problem and needs to be unloaded, but there is no spent fuel pool?**

   Answer

   There are currently no NRC requirements that prohibit decommissioning of reactor support facilities while spent fuel is being stored at the ISFSI. NRC [in 10 CFR 50.54(bb)] requires licensees to submit, for NRC review and approval, a program for managing and providing funding for management of spent fuel following permanent cessation of operation of the reactor until the time that title to, and possession of, the spent fuel is transferred to the Department of Energy (DOE). In addition, NRC requires [in 10 CFR 72.218] licensees to include in this spent fuel management program, a plan for removal of the spent fuel from the reactor site, and a plan for how the spent fuel will be managed before starting to decommission systems and components needed for moving, unloading, and shipping the spent fuel.

   Dry spent fuel storage is safe and environmentally sound. Over the past 20 years, there have been no radiation releases which have affected the public, no radioactive contamination, and no known or suspected attempts to sabotage spent fuel casks or ISFSIs. Dry cask storage systems are designed to withstand earthquakes, floods, tsunamis, tornadoes, projectiles, temperature extremes, and other unusual or accident scenarios, and to continue to perform their safety functions without the need to repackage the fuel into a new system.

   However, because of the uncertainty regarding DOE's program and planning for ultimate spent fuel disposition, the storage period may be longer than initially envisioned. The associated future aging-management inspection or repackaging needs at these sites will continue to be evaluated through the renewal process for licenses and storage Certificates of Compliance. Staff believes that appropriate fuel handling capability or safe movement of the fuel to a site where handling capability is available could be provided in an appropriate timeframe in the event that a problem is identified with a storage cask at a shutdown or decommissioned reactor.

   This issue was included in Petition for Rulemaking by the C-10 group. One of the Petitioner's requests was that NRC require a safe and secure hot cell transfer station coupled with an auxiliary pool as part of an upgraded ISFSI design certification and licensing process. The staff is further considering the issue of fuel handling capability at ISFSIs at shutdown or decommissioned reactors, where there is no spent fuel pool for immediate examination or unloading of the storage casks, as part of our review of that Petition.

5. **How do the waste confidence hearings affect the probability and timing at San Onofre for shipment of nuclear waste to remote interim or permanent storage?**

Answer

The proposed Waste Confidence rulemaking has no impact on the probability or timing of SONGS spent nuclear fuel being shipped to a mined geologic repository because Waste Confidence is the NRC's generic determination regarding the environmental impacts of storing spent nuclear fuel after the end of the licensed life for operations of a nuclear reactor and before final disposal in a repository. The proposed Waste Confidence rulemaking does not authorize the shipment to or disposal of spent nuclear fuel to a mined geologic repository. Similarly, Waste Confidence has no impact on SONGS spent nuclear fuel being shipped to a remote interim storage site because the proposed rulemaking does not authorize the shipment to or storage of spent nuclear fuel at an interim storage facility.

Emergency Preparedness During Decommissioning

Key Messages

1. Emergency preparedness requirements are not being eliminated but aligned to the risk to the public, which is primarily associated with the spent fuel pool, rather than an operating power reactor.

2. Licensese must maintain an onsite emergency plan providing for the classification of emergencies, notification of offsite government authorities, and coordination of offsite organizations responding onsite (i.e., firefighting, medical assistance, etc.).

3. Licensees will be required to maintain the capability to mitigate a spent fuel pool event.

No.	Question	Response
1	Are the current NRC regulations for emergency preparedness for nuclear power plants designed for plants that are decommissioning?	There are no explicit regulatory provisions distinguishing Emergency Preparedness (EP) requirements for a nuclear power reactor that has been shut down from those for an operating power reactor. The EP requirements of 10 CFR 50.47, "Emergency plans," and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50 continue to apply to a nuclear power reactor after permanent cessation of operations and removal of fuel from the reactor vessel. Consequently, to modify their emergency plans to reflect the reduced risk to public health and safety from power reactors that have been permanently shut down, power reactor licensees transitioning to a decommissioning status seek exemptions from certain EP regulatory requirements before amending their emergency plans.
2	What is the scope of exemptions to EP requirements being considered?	The exemptions requested are consistent with those previously granted for decommissioning power reactors and bring the EP requirements for these decommissioning facilities in line consistent with the current requirements for independent spent fuel storage installations (ISFSIs) – 10 CFR 72.32(a). These exemptions include the requirement for formal offsite radiological emergency preparedness (REP) plans and a scaling back of the licensee's onsite emergency response commensurate with reduced risk. The licensee will still be required to maintain an onsite emergency plan addressing the classification of an emergency, notification of designated offsite government officials, and coordination onsite the response onsite of offsite response organizations (i.e., firefighting, medical assistance).

No.	Question	Response
3	What does the elimination of formal offsite radiological emergency preparedness (REP) plans mean? How would actions be taken to protect the public around the facility?	A formal offsite radiological emergency plan requiring certification by FEMA is not required because the risk to public health and safety is expected to be significantly less than when the plant was operating. As such, the requirement for detailed pre-planning within 10-miles (plume exposure pathway) of the facility would no longer exist. In the highly unlikely case where may become necessary, there should also be sufficient time for State or local governments to undertake offsite protective measures. State and local governments have comprehensive emergency management (all hazard) plans in place to address a wide-range of non-radiological emergencies. There is amble data to show that offsite response organizations can effectively carry out protective measures (including evacuations) for a wide range of severe events.
4	If not, how does the NRC determine what emergency preparedness requirements are appropriate to the risk presented by the facility?	When a licensee submits a request for an exemption from EP requirements for an operating nuclear power plant, the NRC staff reviews the request and its associated justifications. The justifications include technical analyses of the type and risk of accidents that could occur at the plant. These analyses are used to provide the NRC reasonable assurance that in granting the requested exemption: (1) an offsite radiological release would not exceed the U.S. Environmental Protection Agency protective action guides at the site boundary (as applicable to design basis accidents such as fuel handling or radioactive waste processing accidents); and (2) in the unlikely event of a severe beyond design-basis accident resulting in a loss of all cooling, sufficient time would exist to initiate appropriate mitigating actions, and if needed, implement offsite protective actions using a comprehensive emergency management plan to protect the health and safety of the public. If the impacts were determined not to require offsite protective actions, then the licensee would not need to maintain compliance with certain offsite EP requirements. For the onsite risks at the plant, the NRC would require that the licensee continue to comply with the appropriate EP requirements.

No.	Question	Response
5	What is an exemption, and why is it required?	An exemption applies to a licensee seeking regulatory relief (no longer needs to comply with regulation). Current EP regulations do not take into consideration reduced risks associated with potential accidents that may occur at a nuclear power reactor that has permanently ceased operation and transferred fuel from the reactor vessel to a spent fuel pool, which presents less risk in its current state than does an operating plant because certain types of accidents are no longer possible. Historically, exemptions to EP requirements have been used to grant regulatory relief on a case-by-case basis. Until an exemption is granted, onsite and offsite EP programs must be maintained and all requirements met.
6	How does the NRC exemption process work, and why are exemptions allowed?	The practice of considering exemptions is a well-established part of the NRC's regulatory process that allows licensees to address site-specific situations or implement alternative approaches for circumstances not necessarily contemplated in the regulations for operating reactors. The exemption process is not unique to decommissioning licensees or to the specific EP technical area, but is an important tool that allows the agency to provide appropriate regulatory relief, permitting licensees to make appropriate modifications to their programs commensurate with the site-specific risks that are present for a permanently shutdown reactor during decommissioning. The NRC makes decisions on exemption requests on a site-specific, case-by-case basis, following an established process that includes the NRC staff's assessment of a detailed technical safety analysis submitted by the licensee. The NRC may grant exemptions from various regulatory requirements in response to a request from a licensee, applicant, or interested stakeholder. For the Commission to consider granting an exemption, the request must provide information that demonstrates that specific criteria in the regulations, specifically those in 10 CFR 50.12, "Specific exemptions," will be met. The NRC staff's review of requests for exemptions pursuant to 10 CFR 50.12 considers whether the exemptions are authorized by law, will not present undue risk to the public health and safety, and are consistent with the common defense and security, and whether any special circumstances, as defined in the regulation, exist. The special circumstances listed in 10 CFR 50.12 reflect some of the reasons that exemptions may be necessary. Special circumstances are present whenever application of the regulation conflicts with other rules or regulations,

No.	Question	Response
		application of the regulation in the particular circumstance would not serve the underlying purpose of the rule, compliance with the regulation would result in undue hardship, the exemption would result in benefit to the health and safety of the public, the exemption would provide temporary relief from the applicable regulation and the licensee has made good faith efforts to comply, or there is present a material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant the exemption. The NRC will grant an exemption only if it concludes that all the required criteria are met. If granted, the exemption, which contains the staff's safety analysis and conclusions, is published in the Federal Register.
7	Will the NRC ensure that adequate safety requirements are in place for power reactors during decommissioning?	Yes, the NRC will continue to maintain appropriate safety requirements, and provide appropriate oversight for all permanently shut down nuclear power reactors through all phases of the decommissioning process. The licensees are required to comply with the existing conditions and technical specifications of their licenses until and unless those requirements are changed via a license amendment or exemption. During decommissioning, licensees typically request license amendments and exemptions from certain regulations, accompanied by a technical justification, to change the requirements based on a facility's permanently shutdown and defueled condition. The NRC staff applies the same rigor in its regulatory and technical evaluation of amendment and exemption requests regardless of whether a facility is in operation or a permanently shutdown and defueled condition. The NRC will only approve a license amendment or grant an exemption if the staff concludes its issuance will continue to provide reasonable assurance of adequate protection of the public health and safety. In addition, the NRC's oversight and inspection activities will continue at all decommissioning sites throughout the decommissioning process.

No.	Question	Response
8	Why is Commission approval required?	In a Staff Requirements Memorandum (SRM) dated May 19, 2008, to SECY-08-0024, the Commission directed that "The staff should request Commission approval for any reduction in the effectiveness of a licensee's emergency plan that requires an exemption from the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50."
9	Does NRC have the ability to impose condition as part of any exemption request that it might approve?	Yes, the NRC staff may grant an exemption based on a requirement that the licensee satisfy certain conditions. In this case, the conditions would be fully described and evaluated, and incorporated into the license when the exemption is granted.
10	The use of the license exemption process appears to undermine the NRC Safety Regulations in that it does not allow meaningful public participation, even though the result may be perceived as resulting in serious compromise to public safety. Why isn't the license amendment process used, instead of the exemption process?	The NRC regulations in 10 CFR Part 50.12 specify conditions under which licensees can seek exemptions from certain regulatory requirements. In certain situations, licensees can also seek amendments.
11	Why has the NRC never refused a request of a decommissioning reactor licensee to be exempted from off-site emergency preparedness requirements?	The NRC staff has not approved all exemption requests as originally submitted. Some licensees have modified or withdrawn their exemption requests in accordance with determinations made during the staff's review. The NRC staff reviews requests for exemptions for decommissioning plants using the same process and standards it uses to review requests for exemptions submitted by licensees of operating plants. The NRC review of exemption applications considers whether the exemption is authorized by law, whether the exemption would present an undue risk to the public health and safety, whether the exemption is consistent with common defense and security, and whether special circumstances, as defined in the NRC's regulations, exist. If the NRC staff determines that a proposed exemption request does not satisfy these requirements, the NRC staff typically provides the licensee an opportunity to supplement its application with additional information. If it becomes apparent through interactions with the NRC staff that the exemption will not be granted,

No.	Question	Response
		the licensee often chooses to withdraw the exemption request, rather than having NRC issue a denial. As a recent example, on August 27, 2013, Southern California Edison (SCE) withdrew its request for an exemption to defer performing the biennial NRC evaluated exercise of its onsite and offsite emergency plans for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3, as required by 10 CFR Part 50, Appendix E, Sections IV.F.2.b and c. As indicated in SCE's letter, the request was withdrawn "[b]ased on discussions with the NRC staff and after consulting with the offsite agencies." SCE requested the exemption on August 8, 2013, shortly after submitting certifications that it had permanently shut down and defueled SONGS Units 2 and 3.
12	When does the FEMA oversight of offsite emergency preparedness, as established in 44 CFR 350, cease?	When approval of the EP Exemption SECY Paper is granted by the Commission, and all required safety evaluation reports have been completed for the NRC's approval of exemption and license amendment requests for the proposed permanently defueled emergency plan (PDEP) and EALs, the NRC will notify FEMA that formal radiological emergency preparedness (as defined in 44 CFR 350) is no longer required as a condition of the Part 50 license.
13	What is the NRC process for reviewing and making decisions on Permanently Defueled Emergency Plan (PDEP) submittals? Please provide the staff and Commissioner processes, as well as accurate and realistic timeframes under which the PDEP is reviewed, considered and available to the public, and voted on.	The NRC staff uses the same process for reviewing and making decisions on permanently defueled emergency plans as the staff uses for reviewing and making decisions on an operating facility's emergency plans. Licensees may make changes to the emergency plan without NRC approval only if the change does not reduce the effectiveness of the emergency plan, and the emergency plan, as changed, continues to meet the requirements in Appendix E to 10 CFR Part 50 and the planning standards of 10 CFR 50.47(b). If the licensee determines that the change involves a reduction in plan effectiveness, the licensee must submit the proposed emergency plan change for prior NRC approval as a license amendment request under 10 CFR 50.90. The NRC review process is initiated when a licensee submits a license amendment request to the NRC Document Control Desk. An acceptance review is performed by NRC staff, to: (1) determine if documentation submitted is sufficient to support the staff's technical review; (2) coordinate necessary

No.	Question	Response
		interfaces with various NRC offices; and (3) establish a schedule for the technical review of the request. As soon as practicable, following completion of the acceptance review, the NRC will publish a notice in the Federal Register. The Federal Register notice will describe the proposed change, describe the procedure for providing comments on the proposed change, announce the opportunity to request a hearing, and provide instructions for requesting a hearing. As part of the review of the license amendment request, the NRC staff verifies that the licensee has informed the appropriate State officials of the proposed change. During the review process, requests for additional information may be issued to the licensee for the purpose of enabling the staff to obtain all relevant information needed to make a regulatory decision on a license amendment request that is well-informed, technically correct, and legally defensible. When the NRC completes its technical review of the proposed change, in the form of a written safety evaluation report, it publishes a notice in the Federal Register indicating the approval or denial of the request. The NRC has a general goal of completing action on a license amendment request within one year; however, staff action on a license amendment request may be expedited to reduce time to approximately 6-9 months based on the request's safety significance, complexity, and assigned priority. If the licensee determines that the plan as changed would not meet the requirements in Appendix E to 10 CFR Part 50 and the planning standards of 10 CFR 50.47(b), the licensee may submit an exemption request, rather than a license amendment request, prior to implementing any changes to its plan. PDEP changes submitted for prior NRC approval, including those that implement exemptions already approved by the Commission, are approved by the Director of the Office of Nuclear Reactor Regulation and do not require separate Commission approval.

No.	Question	Response
14	Under an approved PDEP, does the licensee retain responsibility for onsite emergency response and communication with offsite organizations?	Yes. At all times, the licensee is primarily responsible for safety of the licensed activities, including on-site emergency response and all aspects of coordination and communication with offsite organizations (including, but not limited to, the relevant law enforcement and emergency management organizations). Following approval of the PDEP, the requirement in 10 CFR 50.54(q) to maintain an onsite emergency plan that meets the planning standards of 10 CFR 50.47(b) and the requirements of Appendix E to 10 CFR Part 50 (unless exempted by the Commission), remains in effect. The onsite emergency plan will outline on-shift and augmented organization staffing capable of responding to emergency events appropriate for a reactor that has permanently ceased operation and transferred fuel from the reactor vessel to a spent fuel pool or dry cask storage. The onsite emergency plan will also require the notification of designated offsite government authorities upon the declaration of an emergency event and provide for the coordination of offsite response organizations (e.g., firefighting, medical transportation) onsite.
15	Do you believe a defueled reactor poses essentially the same off-site radiological emergency risk profile as an operating reactor? Why?	No. When a nuclear power plant permanently ceases operations and the licensee defuels the reactor, the risk to the public from an accident drops significantly, since the accident sequences that dominated the operating plant risk are no longer applicable. The primary remaining source of risk to the public is associated with potential accidents that involve the spent fuel stored in the spent fuel pool. Moreover, the predominant design-basis accident for a defueled reactor is a fuel handling accident. The risk of a radiological release from a spent fuel pool at a decommissioned plant would typically be lower than from a spent fuel pool at an operating facility. This is because the heat generated by spent fuel significantly decreases over time following transfer of the spent fuel from the reactor to the pool as fission products decay. In contrast, the amount of heat generated in a spent fuel pool at an operating reactor does not significantly decrease with time because of the additional heat discharged from newly transferred spent fuel as the spent fuel is removed from the reactor and placed in the spent fuel pool every 18 to 24 months.

No.	Question	Response
16	Since the NRC staff studied the potential for fires to occur in spent fuel pools, what is their conclusion on the actual probability of such an event occurring?	Over the past several decades, the NRC has periodically evaluated the safety of spent fuel pools, and has consistently concluded that spent fuel pools are robust structures that are likely to withstand severe earthquakes and other credible challenges and thus provide for safe storage of spent nuclear fuel. Two recent evaluations, including NUREG-1738, “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants” and NUREG-2161, “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor,” have demonstrated that the risks of a radiological release form a spent fuel pool were very low.
17	Has the NRC performed an analysis of the risks of a spent fuel pool fire? If so, what did the NRC conclude?	Over the past 35 years, the NRC has sponsored a number of studies to evaluate various aspects of spent fuel pool safety, security, and risk. A summary of the past studies (including NUREG-1738 for decommissioning plants) is provided in SECY-13-0112, “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor,” which is commonly known as the Spent Fuel Pool Study. A regulatory analysis that analyzed spent fuel pools across the U.S. fleet of nuclear power plants was submitted to the Commission in COMSECY-13-0030. These studies have shown that spent fuel storage is safe and the risk of a release of radionuclides due to an accident is low. The analysis included a review of the risk of a release from a spent fuel pool fire. The results of NUREG-1738 indicated that the risk is low and well within the Commission's quantitative health objectives. The risk was found to be low because of the very low likelihood of a zirconium fire, even though the consequences from a zirconium fire could be serious. The results of more recent studies, such as NUREG-2161, are consistent with the earlier conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking and exposing the fuel, which could lead to a fire.

No.	Question	Response
18	There are 12 sites where civilian nuclear power generating activities have permanently ceased and all that does, or will remain at these facilities is the spent fuel and reactor generated Greater-Than-Class C Wastes stored on site awaiting removal by the Department of Energy. After these facilities permanently ceased operations, each submitted a PDEP to modify parts of the Emergency Plan based on the significantly reduced risks of a radiological event at a decommissioning plant. After these facilities were given approval to make the changes authorized under the PDEP, has there been an event, emergency or threat including natural disasters or hostile acts that proved to be too significant or challenging to be handled by the NRC approved PDEP?	There have been no events, emergencies, or threats, including natural disasters or hostile acts that have proved to be either too significant or challenging to be handled in accordance with the NRC-approved PDEP. The radiological risks are greatly reduced at a decommissioning nuclear power plant and have a very low probability of causing an event that could have impacts offsite. In addition, the sites are required to maintain an NRC-approved security plan to ensure potential hostile threats will be safely and securely managed to protect the plant.

No.	Question	Response
19	NUREG-1738 states that "SFP fires could have health effects comparable to those of a severe reactor accident.... Large seismic events that fail the SFP are the dominant contributor [to causing an SFP fire]." Please confirm the NRC position regarding this statement.	The Commission stands by the finding that the offsite health impacts of a spent fuel pool zirconium fire as evaluated in NUREG-1738 can be comparable to those from a severe accident at an operating reactor as evaluated in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." More recent studies, such as NUREG-1935, "State-of-the-Art Reactor Consequence Analyses" (SOARCA) Report, and NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," have concluded that, for the types of accidents examined at both reactors and spent fuel pools, probability-weighted health impacts (i.e., calculated risk) would be low and well within the Commission's safety goals. In the unlikely event of an accident at either a reactor or spent fuel pool, the probability-weighted offsite economic impacts could be comparable. This is because while the economic impacts of a SFP fire could be very large, a spent fuel pool accident is expected to be less likely than a reactor accident. During the development of NUREG-2161, the NRC staff did not find any new information to challenge the view expressed in NUREG-1738 and earlier studies that large seismic events are the largest contributor to the likelihood of having a large radiological release from the spent fuel pool.
20	[NUREG-1738] "Further, the analysis indicates that timely evacuation, implemented through either pre-planned or ad hoc measures, can significantly reduce the number of early fatalities due to a zirconium fire." Please confirm the NRC position regarding this statement.	The Commission stands by the finding that an early evacuation (which is defined in NUREG-1738 as an evacuation that is initiated and completed before the spent fuel pool release), is effective at reducing the number of early fatalities, because early fatalities arise from very high acute radiation exposures. Large acute radiation exposures can be significantly reduced, if not eliminated altogether, by an evacuation that is completed before the spent fuel pool release begins. As stated in this study, the effect of timely evacuation is the same whether it is implemented through pre-planned evacuation or whether it is implemented through effective ad-hoc measures. NUREG-1738 states that "the overall low risk in conjunction with differences in dominant sequences relative to operating reactors, results in a small change in risk at a decommissioning plant if offsite planning is relaxed."

No.	Question	Response
		The NRC staff continues to believe that, in the unlikely event of a severe beyond design-basis accident resulting in a loss of sufficient water and natural air cooling within the pool, there is sufficient time for offsite agencies to take protective measures under a comprehensive emergency management (all-hazards) plan to protect the health and safety of the public.
21	[NUREG-1738] "[T]he long-term consequences of an SFP fire may be significant. These long-term consequences (and risk) decrease very slowly because Cesium-137 has a half-life of approximately 30 years." Please confirm the NRC position regarding this statement.	The Commission stands by the finding, as stated in NUREG-1738, that as long as a zirconium fire is possible, the long-term consequences of a spent fuel pool fire may be significant. NUREG-1738 illustrates the change in health consequences from a zirconium fire as a function of time since shutdown, and demonstrates that latent fatality risks and long-term collective population doses, which are more sensitive to the inventory of the longer-lived cesium-134 (half-life of two years) and cesium-137 (half-life of 30 years), drop slowly. However, the NRC also considers calculated risk in its consideration, which is the probability of a release multiplied by the radiological consequences of a release (i.e., probability-weighted). This consideration is illustrated by the finding in NUREG-1738 that “the risk at decommissioning plants is low and well within the Commission's safety goals. The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious.” Although the staff cannot completely rule out large radiological releases from a spent fuel pool, the staff’s analysis shows that the probability of a release from a spent fuel pool at a plant in decommissioning decreases with the passage of time due to the drop in decay heat from the spent fuel stored in the pool, and the fact that there would be no further additions to the inventory of fresh spent fuel because the plant is no longer operating.
22	[NUREG-1738] "[T]he consequences from a zirconium fire could be serious." Please confirm the NRC position regarding this statement.	This statement is correct. Over several decades of research, the NRC staff has consistently found that spent fuel pool fires are very high-consequence, but very low probability events.

No.	Question	Response
23	[NUREG-1738] "Insurance, security, and emergency planning requirement revisions need to be considered in light of other policy considerations, because a criterion of "sufficient cooling to preclude a fire" cannot be satisfied on a generic basis." Please confirm the NRC position regarding this statement.	The Commission stands by the finding in NUREG-1738, which was appropriate, and notes that further staff review of safeguards provisions at decommissioning plants has occurred since that document was issued. The NRC has taken a number of additional actions related to this finding. On June 4, 2001, the NRC staff submitted a policy paper "Policy Issues Related to Safeguards, Insurance and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools," SECY-01-100, to the Commission for consideration. The paper provided the Commission with the staff's assessment of the policy implications of the NUREG-1738 study related to decommissioning exemptions for insurance, EP, and safeguards. The paper also recommended the NRC implement Commission policy direction in response to the paper in a future decommissioning rulemaking. Although SECY-01-0100 did not result in a rulemaking regarding insurance, EP, and safeguards requirements for decommissioning plants, it provides the staff's assessment of the findings of NUREG-1738 at that time. The NRC staff is using this assessment, as well as additional information gained through subsequent studies and Commission policy decisions, to inform its review of site-specific license amendment and exemption requests from the recently shut down power reactors on a case-by-case basis. On January 10, 2014, the staff issued for public comment a draft Interim Staff Guidance document, "Draft Interim Staff Guidance on Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants," NSIR/DPR-ISG-02. The guidance will assist NRC staff in processing requests for exemption from EP requirements for nuclear power reactors that are undergoing decommissioning. It considers historical experience and precedent with previously issued exemptions and a number of related studies, including NUREG-1738, SECY-01-0100, and SECY-13-0112. The NRC staff is also considering insights from NUREG-1738 and SECY-01-0100 in conjunction with Commission policy decisions made in response to SECY-93-127, "Financial Protection Required of Licensees of Large Nuclear

No.	Question	Response
		Power Plants During Decommissioning," and SECY-04-0176, "Exemption Requests to Reduce Liability Insurance Coverage for Decommissioning Reactors after Transfer of All Spent Fuel From a Spent Fuel Pool to Dry Cask Storage," in its review of financial protection and insurance exemption requests from decommissioning licensees. In response to SECY-93-127, the Commission approved reductions in the amount of financial protection and insurance required of decommissioning reactor licensees. In response to SECY-04-0176, the Commission approved the denial of exemption requests from decommissioning reactor licensees that requested additional reductions of insurance requirements after transfer of all spent fuel from the spent fuel pool to a dry cask storage. If granted, the exemption requests would have lowered the level of liability insurance required below the minimum levels previously established by the Commission.
24	[NUREG-1738] Figure 2.1 shows that even after a reactor had been shut down for 1 year, it would take only about 3 hours for PWR reactor fuel to heat to 900 degrees Celsius and only about 7 hours for BWR reactor fuel to heat to 900 degrees Celsius, even when the spent fuel pool accident does not prevent the assemblies from being air cooled. Please confirm the NRC position regarding this statement.	The Commission stands by the analysis in NUREG-1738, which is correct given the hypothetical assumptions upon which the analysis is based. Because of its intended purpose (e.g., exemption requests from NRC requirements for offsite emergency preparedness for decommissioning reactors), the staff purposely introduced conservative assumptions into the analysis. These conservatisms include simplified treatment of the thermal-hydraulic response (e.g., cooling of fuel and temperature change) and the use of assumed and often bounding configurations that do not allow for thermal radiation between high powered bundles and low power bundles (as stated in NUREG-1738) and also from the spent fuel assemblies to the spent fuel pool wall liner. In a more realistic calculation, as demonstrated in the recent Spent Fuel Pool Study (NUREG-2161), thermal radiation heat transfer (in addition to air cooling) can play a significant role. For example, it could take more than 10 hours for the fuel to heat up to 900°C even one month after being moved from the reactor to the pool if the assemblies most recently removed from the reactor (i.e., the hottest) are distributed among older, cooler, fuel assemblies. In addition, NUREG-1738 makes simplifying assumptions regarding the pool failure leakage rate that results in instantaneous drain down of the pool. In the Spent Fuel Pool Study, even for a moderate leak scenario, it took more than two hours for the water level to reach the top of the fuel.

No.	Question	Response
		In summary, the analyses performed in NUREG-1738 were based on intentionally conservative assumptions, and resulted in conservative consequences, which was an appropriate regulatory approach for the issue under consideration. More recent and realistic analyses show that the consequences of these accident scenarios are not as severe as assumed in NUREG-1738 and that they take place over a much longer period of time.
25	[NUREG-1738] Figure 2.2 shows that for PWR reactor fuel that is subject to a spent fuel pool accident that does not allow for air cooling to occur (the so-called adiabatic case in which the pool would only partially drain and thus preclude air circulation), it would take only 6 hours for the fuel to heat up even one year after the reactor shuts down. Please confirm the NRC position regarding this statement.	The Commission stands by the analysis in NUREG-1738, which is correct given the hypothetical assumptions upon which the analysis is based. However, as stated above, there are conservatisms associated with the analysis in that document. In an adiabatic calculation, both the oxidation energy and radiation heat transfer are not taken into account. While the oxidation energy tends to increase the fuel temperature, thermal radiation would limit the fuel heatup. However, for partial drain down cases, the blocked airflow can limit the more energetic air (as opposed to steam) oxidation reaction, while thermal radiation only depends on the temperature and would play an important role in limiting the fuel heatup rate. The Spent Fuel Pool Study (NUREG-2161), which – unlike NUREG-1738 – addressed these effects, showed that for small leak scenarios with blocked air flow at 107 days after shutdown for a specific site, it would take more than 10 hours to increase the fuel temperature to 650°C. For an adiabatic calculation, the actual time for the hottest fuel assemblies to reach 900°C requires a plant specific calculation.
26	What are the risks for a spent fuel pool accident compared to that for an operating power reactor?	While a SFP would normally contain more fuel assemblies than would be present in a single operating reactor core, without the diving force of high temperatures and pressures present in an operating reactor, there are fewer potential accident scenarios which could lead to a significant offsite radiological release that exceed the U.S. Environmental Protection Agency's (EPA) Protective Action Guidelines (PAGs). As the spent fuel ages, its ability to generate heat diminishes due to the decay of fission products. Additionally, unlike operating reactor accident sequences that could lead to a large early release, accident scenarios at decommissioning plant SFPs evolve slowly and would provide adequate time to initiate mitigation or protective actions.

No.	Question	Response
27	A document entitled, “Environmental Impacts of Storing Nuclear Fuel and High-Level Waste from Commercial Nuclear Reactors: A Critique of NRC’s Waste Confidence Decision and Environmental Impact Determination,” authored by Dr. Gordon Thompson, describes SFP fires that could start in as little as 2 or 3 hours after cooling is lost to uncovered fuel. This appears inconsistent with criteria used by the NRC. Please clarify.	Based on the results of various studies, the NRC believes that the probability of a zirconium fire in the SFP is very low and timely implementation of designated mitigation measures are available to address initiating conditions that could lead to a zirconium fire in the SFP. In the unlikely event that mitigation measures could not be implemented in a timely manner to preclude the possibility of a zirconium fire in the SFP, the staff believes that sufficient time would exist for the implementation of offsite protective actions using comprehensive emergency management plans (CEMPs) to provide for the health and safety of the public. Three such studies are: NUREG/CR-6451, “A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants” (ADAMS Accession No. ML082260098); NUREG-2161, “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor” (ADAMS Accession No. ML14255A365); and NUREG-1738, “Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants” (ADAMS Accession No. ML010430066). Regarding a fire that could start in as little as 2 or 3 hours, the NRC staff notes that this could be true for spent fuel 1 to 3 months after removal from the reactor. However, decommissioning reactors, with fuel in the pool that is much older, would take much longer to heat up. The exemption requests must include site-specific analyses demonstrating that at least 10 hours is available from the time there is a total loss of all cooling to the fuel until fuel temperature reaches 900 degrees Celsius.
28	What basis is there for assuming reasonable assurance that there will be time for preventative and mitigation actions?	The risk associated with a zirconium fire event is directly related to decay heat from the fuel, and therefore, the time since shutdown. NUREG-1738 conservatively estimated that greater than 100 hours would be available before SFPs lowered to within 3 feet of the top of the fuel for loss of cooling events when pressurized water reactor (PWR) fuel has decayed at least 60 days. The NRC currently requires, and will continue to require for decommissioning power reactors, that the equipment, procedures, and trained personnel necessary be designated in the exemption request to provide alternate spent

No.	Question	Response
		fuel cooling in the unlikely event of a SFP drain down. An example would be the licensee's retention of the license condition to maintain the requirements of 10 CFR 50.54(hh)(2) pertaining to SFP mitigation actions. Additionally, the NRC has studied evacuations nationwide and found local responders are capable of implementing protective actions for the public. A study conducted by Sandia National Laboratories, NUREG/CR-6864, "Identification and Analysis of Factors Affecting Emergency Evacuations" (ADAMS Accession No. ML050250245), concluded that "evacuations successfully protect public health and safety over a broad range of initiating circumstances and challenges" and "large-scale evacuations in the United States, whether preplanned or ad hoc, are very effective and successfully save lives and reduce the potential number of injuries associated with the hazard." The SFP fire scenario would be a relatively slow developing event as compared to incidents many local responders routinely respond to. It is important to recognize that the regulations will still require the licensee to maintain in effect its onsite radiological emergency response plans and that the local and State governments continue to maintain the responsibility of protecting citizens in the event of emergencies and, if needed, would implement protective actions using the CEMP process.
29	How has the NRC demonstrated that a licensee of a decommissioning reactor would be able to mitigate the potential consequences of an accident at a spent nuclear fuel pool within ten hours if the accident was caused by a severe, unexpected initiating event, such as a massive and devastating earthquake or a terrorist attack?	Decommissioning power reactor licensees are required to have and maintain effective emergency plans that meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. Licensees that have proceeded to decommissioning an operating reactor have submitted requests from portions of these regulatory requirements. The NRC staff evaluates site-specific analyses supporting the licensee's request for exemptions from emergency plan requirements. The analyses would be used to provide the NRC reasonable assurance that in granting the exemption: (1) an offsite radiological release would not exceed the U.S. Environmental Protection Agency protective action guides at the site boundary for design basis accidents applicable to shutdown reactors, and (2) sufficient time would exist to initiate appropriate mitigating actions or offsite protective actions, if needed, to protect public health and safety in the unlikely event of a

No.	Question	Response
		severe beyond design-basis accident resulting in loss of sufficient cooling in the pool. In normal operations, the spent fuel pool is filled with water to keep the spent fuel cool. The staff has generally determined that 10 hours is a conservative minimum time available to implement mitigation actions, and/or initiate protective offsite measures using a State and local government's comprehensive emergency management plan, if the water were to drain from the pool and jeopardize the ability to keep it sufficiently cooled. However, in a hypothetical spent fuel pool accident scenario, 10 hours is not the expected amount of time it would take for water to drain from the pool. A beyond design-basis accident that results in the water draining from the pool (whether a full or partial drain-down) would likely take much longer than 10 hours because of the robust construction of the spent fuel pool and the large volume of water in the pool. Furthermore, particularly for older fuel, air cooling and other heat removal mechanisms following loss of cooling water may be sufficient to keep the fuel cool indefinitely or would significantly extend the fuel heat-up time. To be conservative, the exemption analysis and 10-hour criterion for mitigating the potential consequences pool do not credit natural air cooling and water cooling in the spent fuel pool after the event. Instead, it is assumed that the fuel immediately begins to heat up without any natural removal of its energy. The NRC staff accepts this simplified approach and time estimate in making its regulatory decisions whether to grant the exemption and if granted, when an exemption becomes effective (e.g., typically 12-24 months after shutdown). The NRC staff reviews the analyses provided by the licensee to verify that a minimum of 10 hours is still available to restore cooling, or implement offsite protective measures, before the fuel slowly heats up to a temperature that could cause a zirconium fire. A licensee may in part rely upon the established capabilities it had for its operating reactor as required in 10 CFR 50.54(hh)(2) to mitigate the potential consequences of a severe accident at a spent fuel pool.

No.	Question	Response
		In addition to normal plant structures, systems and components to maintain water levels and cooling to the spent fuel, there are redundant and independently powered equipment to perform these functions located onsite for operating reactors. Decommissioning licensees may choose to retain some of these capabilities to provide assurance they can implement mitigation measures within 10 hours, or may commit to another strategy that provides similar assurance.
30	What is the purpose of the interim staff guidance on emergency planning exemptions issued by the NRC?	The NRC has developed draft interim staff guidance (ISG) entitled, "Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants," to address licensee exemption requests from certain NRC emergency planning (EP) requirements. The draft ISG provides a technical discussion and an overview of existing guidance for reviewing emergency planning exemption requests, as discussed above. In developing this guidance, the NRC relied upon its previous exemption review experience. Additionally, the guidance was informed by existing studies on spent fuel pool fires and risks. The staff is planning to publish a final ISG, following incorporation of public comments, as appropriate, and experience gained from its review of current exemption requests.
31	The NRC's ISG on emergency planning would allow exemption from offsite emergency planning requirements for decommissioning reactors 12 months after permanent shutdown. This is inconsistent with previous NRC staff statements that EP exemptions are evaluated on their technical merits on a case-by-case basis. Please clarify.	The staff has revised the proposed ISG to clearly state that an exemption request requires a site-specific analysis to define the date when exemptions could be permitted. The required analysis must show that at least 10 hours is available from the time that both water and air cooling is lost to the spent fuel until the hottest fuel assembly reaches 900°C to take mitigating actions, or if required, to take protective actions using a CEMP approach. The analysis contains the conservative assumption that the spent fuel immediately begins to heat up without any heat from the fuel being transferred to the surroundings. In reality, there would be heat loss to the air surrounding the fuel, to any remaining water in the pool or on fuel surfaces, to any nearby, cooler fuel assemblies and to the SFP walls. As discussed in NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," radiative heat transfer (in addition to air cooling) can play a significant role. For example, it could take

No.	Question	Response
		more than 10 hours for boiling water reactor fuel to heat up to 900°C after 1 month of being removed from the reactor, if the most recently removed fuel is distributed among older, cooler fuel assemblies in the SFP.
32	The NRC's draft ISG states that "after a certain amount of time, the overall risk of a zirconium fire becomes insignificant." Why is there reasonable assurance of an increased probability that the fuel is coolable?	Particularly for older fuel, air cooling and other heat removal mechanisms following loss of cooling water may be sufficient to keep the fuel cool indefinitely or significantly extend the fuel heat up time. Although the staff cannot completely rule out the possibility of a radiological release from a SFP, the probability of a release from a SFP at a decommissioning power reactor decreases with the passage of time due to the drop in decay heat of the fuel stored in the SFP, and the fact that there would be no further additions to the inventory of fresh spent fuel into the SFP, as the plant is no longer operating. More recent analyses have been performed with site-specific information to determine whether a release from a SFP could occur, considering that site's practice regarding the physical arrangement of the fuel in the spent fuel pool. For example, SECY-13-0112 demonstrated that a release is not expected to occur at the operating power reactor site studied for at least 72 hours following a large, beyond design-basis seismic event that occurs more than 60 days after shutdown. In this study, the reference plant SFP contained fuel recently removed from the reactor, which would be more susceptible to a radiological release scenario than would a SFP at a decommissioning power reactor site. NRC studies also show that fuel that has aged sufficiently can be cooled by air or water spray. The requirement for emergency response capability is not based upon the likelihood of an accident, but rather upon defense-in-depth principles that are part of all NRC regulatory actions. Emergency preparedness is a defense-in-depth measure required to mitigate the consequences of very unlikely accidents, regardless of the probability of those accidents. Defense-in-depth is provided by mitigation procedures that provide cooling and makeup capability to the SFP should there be a loss of water inventory.

No.	Question	Response
33	On what specific basis (and using what specific analysis) has the NRC concluded that it would be possible, within ten hours after the initiating event, to: a) repair all key safety equipment; b) repair what could be a large breach in the spent fuel pool; and c) ensure the capability to continually refill the spent fuel pool with fresh water as it drains in the midst of a broader response to or consequences of a severe seismic event (or other natural hazard) or on-going terrorist attack?	The NRC staff has determined that an accident caused by a severe event that results in the water draining from the pool (whether a full or partial drain down) is too unlikely to have been considered in the initial plant design criteria (a beyond-design-basis accident). Such a severe event would likely allow more than 10 hours for mitigation or protective measures because of the robust construction of the spent fuel pool and other features of the pool design that are intended to prevent a rapid loss of water. Furthermore, particularly for older fuel, air cooling and other heat removal mechanisms following loss of cooling water may be sufficient to significantly extend the fuel heat-up time or keep the fuel cool indefinitely. Although the staff cannot completely rule out large radiological releases from a spent fuel pool, the staff believes that the probability of a release from a spent fuel pool at a plant in decommissioning decreases with the passage of time, due to the drop in decay heat from the spent fuel stored in the pool. In addition, because the plant is no longer operating there would be no further additions to the inventory of fresh spent fuel. More recent analyses have been performed with site-specific information to determine whether a release from a spent fuel pool could occur at the site studied, considering that site's practice regarding the physical arrangement of the fuel in the spent fuel pool. In particular, the study transmitted to the Commission with NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," provided consequence estimates of a hypothetical spent fuel pool accident initiated by a low-likelihood seismic event at a reference plant based on the Peach Bottom BWR Mark I spent fuel pool. The study demonstrated that a release is not expected to occur at the plant studied for at least 72 hours following a large, beyond design-basis seismic event that occurs more than 60 days after shutdown. The NRC has determined that it is, therefore, likely that there would be more than 10 hours to mitigate the potential consequences of a severe accident at a spent fuel pool that results in SFP water loss.

No.	Question	Response
		For evaluating exemption requests from NRC requirements for offsite emergency preparedness for decommissioning reactors, the staff purposely introduced conservative assumptions into the analysis. Specifically, the analysis and 10-hour criterion for mitigating the potential consequences of a beyond design-basis accident at a spent fuel pool does not credit the natural air cooling and water cooling in the spent fuel pool after the event, as a modeling simplification. Rather, the analysis assumes that the fuel immediately begins to heat-up without any natural removal of its energy. These conservatisms include simplified treatment of the thermal-hydraulic response and the use of assumed and often bounding configurations that do not allow for thermal radiation between high power bundles and low power bundles (as stated in NUREG-1738) and also from the spent fuel assemblies to the spent fuel pool wall liner. In a more realistic calculation, as demonstrated in the recent Spent Fuel Pool Study (SECY-13-0112), thermal radiation heat transfer (in addition to air cooling) can play a significant role. For example, it could take more than 10 hours for the fuel to heat up to 900°C only one month after being moved from the reactor to the pool if the assemblies most recently removed from the reactor are distributed among older, cooler fuel assemblies. The NRC staff uses the simplified approach to determine the time estimate, as part of its regulatory decision whether to grant the exemption, and to determine when a granted exemption would become effective (i.e., after the fuel has been cooled for a certain period of time, typically 12-24 months after shutdown). The NRC staff reviews the analyses provided by the licensee to verify that a minimum of 10 hours is still available to restore cooling or implement offsite protective measures, once it is assumed that all cooling is lost, and before the fuel heats up to a temperature where rapid oxidation of the fuel cladding could occur, commonly referred to as a zirconium fire. The capabilities specified in 10 CFR 50.54(hh)(2), put in place after the September 11, 2001, terrorist attacks, are one means by which licensees may provide the capability to mitigate the potential consequences of a severe

No.	Question	Response
		accident at a spent fuel pool. Regulatory standards under 10 CFR 50.54(hh)(2) provide that, in addition to normal plant structures, systems, and components to maintain water levels and cooling to spent fuel, redundant and independently powered equipment needed to perform these functions are located onsite. Decommissioning licensees requesting exemptions from emergency preparedness requirements may choose to retain these systems, or commit to another strategy that provides similar assurance. The 10-hour time frame is not intended to be the time it would take to repair all key safety systems or to repair a large breach in the spent fuel pool. Rather, considering the very low probability of beyond-design-basis events affecting the spent fuel pool, in the staff's professional judgment, 10 hours provides a reasonable time period to implement these pre-planned mitigation measures to provide makeup or spray to the spent fuel pool before the onset of zirconium cladding ignition, or, if necessary, to initiate offsite protective measures. The repair of required safety systems or repair of a large breach in the spent fuel pool would be assessed and performed by additional staffing and resources responding to the site as part of the licensee's on-site emergency plan to ensure long-term cooling of the spent fuel.
34	What initiating event could be expected to result in an accident scenario resulting in a loss of SFP water inventory?	The risk of a SFP accident is dominated by a beyond design basis earthquake event, which could challenge SFP liner integrity. Assuming certain storage configuration and heat decay times, spent fuel assemblies could heat up if SFP water inventory is lost. NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," showed the likelihood of a radiological release from the spent fuel after an analyzed earthquake at the reference plant (GE BWR-4 / Mark I Containment) to be about one time in 10 million years. The study considered a severe earthquake with ground motion stronger than the maximum earthquake reasonably expected to occur, which the NRC expects is more challenging for a spent fuel pool structure than that experienced at the Fukushima Daiichi Nuclear Power Plant during the earthquake that occurred off the coast of Japan on March 11, 2011.

No.	Question	Response
35	What other scenarios were considered for spent fuel pool accidents, and what were the results of the associated analyses?	Specific to SONGS, the licensee provided analyses that also considered the following SFP scenarios: • Loss of all heat removal capability – Indicated that as of September 30, 2013, the time to boil is approximately 90 hours and there is an additional 11 days available to restore cooling to the SFP before cooling water level reaches 10 feet above the top of the fuel. • Total loss of cooling water inventory with air cooling – Indicated that as of August 31, 2014, the fuel cladding will not exceed 565°C. • Total loss of cooling water inventory without air cooling – Indicated that the 10 hour threshold would be reached after 31 months of decay time from the time of permanent shutdown (or as of August 2013). • Shine from an empty SFP – Indicated that as of June 12, 2013, the shine from the spent fuel assemblies is well below the acceptance criteria of 100 mrem for a two hour period to a member of the public. • Fuel handling accident – Indicated that the dose analysis results of a fuel handling accident in the Fuel Handling Building is well below the EPA PAGs. Similar analyses were provided by the other decommissioning licensees.
36	Why does the NRC's evaluation of EP exemption requests apparently disregard the risk associated with terrorist attacks and hostile actions?	The licensee's Security Plan must provide high assurance for the physical protection of the SFP. This is the same level of protection that was required during reactor operations. In cases where exemptions are issued for decommissioning reactors, the exemption eliminates the definition for a "hostile action" and its related requirements, but there are elements for security-based events that are maintained. Specifically, the classification of security-based events, notification of offsite authorities, and coordination with offsite agencies under a CEMP approach are still required. The NRC's letter to Congress regarding the National Academy of Sciences "Study on the Security and Safety of Commercial Spent Nuclear Fuel" (ADAMS Accession No. ML050280428), stated in part, that "today, spent fuel is better protected than ever. The results of security assessments completed to date clearly show that storage of spent fuel in both SFPs and in dry storage casks

No.	Question	Response
		provides reasonable assurance that public health and safety, the environment, and the common defense and security will be adequately protected." The design basis threat provides reasonable hypothetical threats for radiological sabotage, to which licensees must be able to respond. In order to assure that this threat statement remains a valid basis for the design of physical protection systems, the staff routinely reviews and analyzes a range of intelligence information. Every 12 months the staff assesses the threat environment for that 12-month period and formally provides its conclusions to the Commission in a report. If significant information were received that called into question the adequacy of the design basis threat statements, the staff would immediately notify the Commission. The NRC staff also continuously engages with the U.S. intelligence community so that it can immediately respond to credible threats to licensees. Requirements for identifying anticipated threats, maintaining preparedness to address those threats, protecting onsite workers, and recommending actions to protect surrounding communities are integral to the regulations. The anticipated threats that licensees must be capable of responding to also include natural phenomena and security events. In the event of a credible threat to the safe and secure operation of the facility, the licensee is required to notify the NRC Operations Center, which is staffed continuously. The NRC staff monitors the situation and the licensee's response to the threat. If conditions warrant, the NRC will activate its Incident Response Center and begin coordinating with other Federal and State agencies.
37	Will the NRC continue to inspect emergency preparedness activities during decommissioning?	After certification of permanent cessation of operation is received by NRC, the facility is no longer subject to Reactor Oversight Process (ROP). However, EP inspection activities will be continued under Inspection Manual Chapter 2561, "Decommissioning Power Reactor Inspection Program." The inspection program is comprised of two major elements: (1) core Inspection, and (2) discretionary Inspection (i.e., reactive and initiative inspections). The NRC inspection program will remain in effect until the license is terminated. In addition, Inspection Procedure (IP) 82501, "Decommissioning Emergency Preparedness Program Evaluation," and IP 82401, "Decommissioning

No.	Question	Response
		Emergency Preparedness Scenario Review and Exercise Evaluation," were created to specifically address inspection activities in the EP arena. Finally, an NRC resident Inspector will typically remain onsite for a period of 6 to 12 months after the 50.82 certification is submitted to the NRC, and will be present for significant decommissioning activities including fuel movement.

Physical Security During Decommissioning

1. Why are decommissioning reactors no longer required to maintain the same physical security requirements they had when they were operating?

Answer

If a licensee submits an exemption request from the physical security requirements for a power reactor, a detailed technical review of the request will be performed to ensure the exemption(s) are authorized by law, will not present an undue risk to the health and safety of the public, are consistent with the common defense and security; and special circumstances, as defined in the NRC's regulations, are present.

Decommissioning nuclear power plants that have ceased operations and no longer have fuel in the reactor core often request exemptions to certain physical security requirements that were intended for operating power plants. However, as long as there is fuel onsite, a decommissioning power plant must continue to maintain a physical protection program that provides high assurance that the activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. The licensee must continue to demonstrate a security strategy to defend against spent fuel pool sabotage scenarios governed by the design basis threat.

2. How does the NRC provide oversight of the security program at decommissioning nuclear power plants?

Answer

The NRC provides oversight of licensee security programs at decommissioning nuclear power plants through a security inspection program that verifies compliance with applicable regulatory requirements. The security inspection program examines licensee activities in order to assess performance and to assure that the licensee's overall security program is meeting the objective of providing high assurance of protection against the design basis threat. The following attributes of a licensee's security program are inspected for decommissioning nuclear power plants: (1) access authorization; (2) access control; (3) equipment performance, testing, and maintenance; (4) protective strategy evaluation; (5) protection of safeguards information; (6) security training; and (7) target sets.

3. What does the term "high assurance" mean for the security of a decommissioning nuclear power plant as compared to "reasonable assurance"?

Answer

The term "high assurance" in regard to the security of a decommissioning nuclear power plant means that the objective of the physical protection program is to prevent a spent fuel sabotage event. Specifically, the program must ensure that the capabilities to detect, assess, interdict, and neutralize threats up to and including the design basis threat of radiological sabotage as stated in 10 CFR 73.1, are maintained at all times. The term "reasonable assurance" means that a defense-in-depth strategy should be used and would imply that various barriers to prevent or mitigate an event could fail (e.g., plant design, operator response/training, etc.).

4. **Does the NRC participate in or observe any on-site security drills performed at decommissioning reactors to verify their security capabilities?**

   Answer

   Yes, the NRC staff participates in or observes onsite security drills performed by the licensee at decommissioning reactors as part of the NRC's ongoing inspection activities to verify the site's security capabilities. Specifically, the NRC's inspection program verifies that the licensee continues to meet the applicable security regulations.

   The NRC inspection program typically includes NRC observation of licensee-conducted force-on-force exercises. Licensee force-on-force exercises assess their ability to defend against the design basis threat for radiological sabotage and provide valuable insights that enable the NRC to evaluate the effectiveness of licensee security programs. They are an essential part of the oversight of the security of these facilities.

   NRC inspectors from Headquarters and the Regional offices monitor licensees' security activities throughout the year. The inspectors provide firsthand, independent assessments of plant conditions and licensee performance, document their findings in writing, and conduct follow-up inspections to ensure that the licensee has made any necessary corrections.

5. **Should there be a potential security threat at a decommissioning reactor, what NRC processes are in place to determine the nature of the potential threat, ensure the capability of each site to address any threats, and to protect the workers and the surrounding communities?**

   Answer

   Through its rulemaking, licensing, and inspection activities, the NRC evaluates the licensee's plans and procedures for identifying and responding to anticipated threats to the safe operation of the facility. The NRC has established regulations and policies that contain design standards and programmatic requirements to address normal operations, anticipated external events, and design basis accidents. The NRC reviews the licensee's plans for compliance with the regulations and policies through its licensing process. The NRC provides oversight of the licensee's implementation of its license requirements through its inspection activities.

   The design basis threat provides reasonable hypothetical threats for radiological sabotage, to which licensees must be able to respond and repel. In order to assure that this threat statement remains a valid basis for the design of physical protection systems, the staff routinely reviews and analyzes a range of intelligence information. Every 12 months the staff assesses the threat environment for that period and formally provides its conclusions to the Commission in a report. If significant information were received that called into question the adequacy of the design basis threat statements, the staff would immediately notify the Commission. The NRC staff also continuously engages with the U.S. intelligence community so that credible threats can be immediately communicated to licensees.

   Requirements for identifying anticipated threats, maintaining preparedness to address those threats, protecting onsite workers, and recommending actions to protect surrounding

communities are integral to the regulations. The anticipated threats that licensees must be capable of responding to include natural phenomena and security events.

In the event of a credible threat to safety and security of the facility, the licensee is required to notify the NRC Operations Center, which is staffed continuously. The NRC staff monitors the situation and the licensee's response to the threat. If conditions warrant, the NRC will activate its Incident Response Center and begin coordinating with other Federal and State agencies. In addition, the NRC will notify licensees of potential threats to their facilities received from other Federal, State, or tribal sources so that licensees can prepare as appropriate.

6. **How will the spent fuel dry casks be secured once all of the fuel has been removed from the pool and placed on the ISFSI pad at decommissioning facilities?**

   Answer

   Once all of the fuel has been removed from the spent fuel pool and placed in storage casks in an Independent Spent Fuel Storage Installation (ISFSI) it will be protected by the requirements within the security orders issued following the terrorist attacks of September 11, 2001, and the requirements in 10 CFR Part 72, *"Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, And Reactor-Related Greater Than Class C Waste*" and 10 CFR 73.55, "*Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage"* as modified by § 72.212 Conditions of general license issued under § 72.210.

   The Code of Federal Regulations provides basic security requirements for ISFSIs, which include well-trained and armed security officers, physical barriers, intrusion detection and surveillance systems, vehicle barriers, and coordination with local response organizations. The NRC also issued security Orders requiring additional security measures for ISFSIs. In order to protect the information from individuals with a malevolent intent, the actual requirements in these security orders are designated at Safeguards Information (SGI) and are not publicly available.

7. **In late May 2014, the Commission released its votes that unanimously accepted the staff's recommendations not to require decommissioning reactors to be subjected to NRC's force-on-force exercises, which are designed to validate the facilities' security capability. What is the basis for not requiring force-on-force exercises?**

   Answer

   The NRC determined that the force-on-force inspections for decommissioning power reactors are not warranted because the current security inspection program, which includes the observation of licensee-conducted force-on-force exercises, provides adequate oversight and verification of the security posture given a reduction in both risk and the number of target sets at decommissioning power reactors.

8. **According to the NRC inspection manual, decommissioning reactors with spent fuel still in the spent fuel pool receive 12-24 hours' worth of security inspections each year. But operating reactors receive well more than 100 hours' worth of security inspections.**

Additionally, NUREG-1738 concluded that "SFP [spent fuel pool] fires could have health effects comparable to those of a severe reactor accident….Large seismic events that fail the SFP are the dominant contributor [to causing an SFP fire]."

If the NRC believes that the consequences of an accident at a spent fuel pool could be equivalent to those of an accident at an operating reactor, why did the Commission assert in its votes that the validation of security measures in place at the spent fuel pool could be met via security inspections when the time spent on those inspections for decommissioning reactors is nowhere near equivalent to the time spent on security inspections for operating reactors?

Answer

NRC's current decommissioning power reactor inspection program was tailored to reflect an inspection protocol for decommissioning reactors with fuel that had aged for many years. After the events of September 11, 2001, the NRC recognized the need to revise the security requirements and inspection programs for new reactors, operating power reactors, and decommissioning reactors. The security inspection activities conducted by the NRC at each of these facilities differ based on the structures, systems, and components that require protection due to the facility's operating status and risk profile. Thus, there are differences within the security inspection programs for new reactors, operating reactors, and decommissioning reactors.

The NRC just completed revising the security inspection program for decommissioning reactors. The revision addressed issues associated with recent and future decommissioning of nuclear power reactors. The revised security inspection program for decommissioning nuclear power reactors will involve a total of 95 hours of security inspections per site, each year. These inspections are designed to ensure that licensees are effectively implementing their physical protection programs consistent with applicable requirements. Within the scope of this overall physical protection program evaluation, the staff will observe and assess licensee-conducted force-on-force exercises.

9. **How do the exemptions from security requirements relate to waste confidence? Does that document assume that all security requirements are waived?**

Answer

The environmental impacts of security exemptions are neither explicitly evaluated nor resolved generically in the Waste Confidence Generic Environmental Impact Statement (GEIS). The GEIS assumes the baseline condition that all licensees will comply with current NRC requirements for security at decommissioning nuclear power plants.

The Waste Confidence analysis does not assume the granting of any security exemptions. The exemptions from NRC security requirements are evaluated on a case-by-case basis and are subject to separate, site-specific safety, security and environmental reviews. The analysis in the Waste Confidence GEIS has no bearing on the granting or denial of a security exemption. The granting of security exemptions does not reduce the security level at a site such that the probability of a successful attack is increased. Consequently, the granting of security exemptions prior to or during the continued storage period does not change the impact determinations in the Waste Confidence GEIS.