April 24, 2014

*VIA ELECTRONIC MAIL*

Andy Imboden, Branch Chief
Communications, Planning, and Rulemaking
Waste Confidence Directorate
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Re: Need for a supplemental waste confidence DGEIS (Docket NRC-2012-0246)

Dear Mr. Imboden:

Through this letter, the State of Vermont, the State of Connecticut, and the Commonwealth of Massachusetts formally request that the NRC Staff prepare a supplemental Waste Confidence Draft Generic Environmental Impact Statement ("DGEIS") in light of recent information and events. The current DGEIS contains many assumptions, including:

1. That high-burnup spent fuel does not present unique problems for long term storage of spent nuclear fuel.

2. That the consequences of a severe spent fuel pool accident are appropriately bounded, including the off-site economic impacts and the time needed for off-site decontamination.

3. That NRC oversight will avoid adverse environmental impacts from unforeseen safety problems and will ensure the development of new technologies when needed.

While Vermont, Connecticut, Massachusetts, and others have already presented extensive comments criticizing these assumptions, this letter provides new and significant information that is not addressed by the DGEIS. Because this information—which was not available before the December 20, 2013 deadline for commenting on the DGEIS—is both new and significant, the NRC Staff is obligated to evaluate it and issue a supplemental DGEIS for public comment.

In *Marsh v. Oregon Natural Resources Counsel*, 490 U.S. 360 (1989), the U.S. Supreme Court held that federal regulations "impose a duty on all federal agencies to prepare supplements to either draft or final EIS's if there are significant new circumstances or information relevant to environmental concerns and bearing on the proposed action or its impacts." 490 U.S. at 372 (quotation omitted). When there remains a major federal action to occur and "the new information is sufficient to show that the remaining action will affect the quality of the human environment in a significant manner or to a significant extent not already considered, a supplemental EIS must be prepared." *Id*. at 374 (quotation omitted). In these situations, it does not suffice to address the new information in the final impact statement. Rather, a supplemental EIS is needed to serve NEPA's action-forcing purpose in two important respects. *See Baltimore*

*Gas & Electric Co. v. Natural Resources Defense Council, Inc.*, 462 U.S. 87, 97 (1983); *Weinberger v. Catholic Action of Hawaii/Peace Education Project*, 454 U.S. 139, 143 (1981). First, a supplemental EIS is needed to ensure the agency can "carefully consider" all available information before making its decision. *Robertson v. Methow Valley Citizens Council*, 490 U.S. 332, 349 (1989). Second, a supplemental EIS is needed so that "the relevant information will be made available to the larger audience that may also play a role in both the decisionmaking process and the implementation of that decision." *Id.*

The NRC has incorporated these well-established principles in the regulations applicable to all environmental impact statements:

> (a) The NRC staff will prepare a supplement to a draft environmental impact statement for which a notice of availability has been published in the FEDERAL REGISTER as provided in § 51.117, if:
>
> ***
>
> (2) There are significant new circumstances or information relevant to environmental concerns and bearing on the proposed action or its impacts.

10 C.F.R. § 51.72(a)(2). The NRC has held that it must prepare a supplemental draft EIS when the new information "present[s] a seriously different picture of the environmental impact of the proposed project from what was previously envisioned." *In re Union Elec. Co.*, CLI-11-05, 74 N.R.C. 141, 167-68 (2011) (quotations and alteration marks omitted).

The new evidence presented here meets that standard. The current DGEIS does not address important information that has arisen since the date of its publication.

**I. New and Significant Information on the Problems of High-Burnup Fuel**

The DGEIS says little about the potential environmental impacts of high-burnup fuel and its storage in spent fuel pools. And what the DGEIS does say is refuted by recent studies and analyses of the impact of storing high-burnup fuel in spent fuel pools.

For example, the DGEIS dismisses the danger of a criticality accident in a spent fuel pool because NRC regulations require plant operators to maintain adequate boron levels to absorb neutrons and prevent criticality:

> Licensees are required to demonstrate that some margin to criticality is maintained for a variety of abnormal conditions, including fuel-handling accidents involving a dropped fuel assembly. The environmental impacts are small, therefore, because criticality accidents in spent fuel pools are prevented.

DGEIS at 4-70. New evidence shows that when high-burnup fuels are used and placed in the spent fuel pools at certain reactors, it can create special problems that interfere with boron control. Ex. 1 (R. Alvarez *The Storage and Disposal Challenges of High Burnup Spent Power Reactor Fuel* (Jan. 3, 2014)) at 9-11. As the DGEIS acknowledges, high-burnup fuel is likely to

remain in spent fuel pools for much longer than the 5 years of normal fuel and possibly as long as 20 years. DGEIS at 2-25. However, that extended time in the pool—combined with the much larger inventory of radionuclides in the high-burnup fuel—places additional demands that require the use of neutron-absorbing panels in the spent fuel pools. Ex. 1 at 6-11. Those panels are subject to deterioration causing a loss of neutron absorption ability and the release of particles into the spent fuel pool. *Id*. at 10. While one can attempt to address this by adding more boron to the water in the spent fuel pool at pressurized water reactors, the boron reacts with the concrete used for the walls of the pools and causes it to be more susceptible to leaks. *Id*. at 11. High-burnup fuel thus requires enhanced chemistry controls and more neutron-absorbing panels. *Id*. But the pools are already densely packed, and the additional equipment in the pools restricts water and air circulation, making the pools more vulnerable to systemic failures from an inability to remove the increased decay heat from high-burnup fuels. *Id*.

NRC contractors, the Electric Power Research Institute ("EPRI"), and the National Academy of Scientists have all raised concerns about high-burnup fuel. Ex. 1 at 2-3. The NRC itself has also recognized that there is inadequate information on the structural integrity of high-burnup fuels after 20 years. Ex. 2 (NRC Division of Spent Fuel Storage and Transportation Interim Staff Guidance-24, Revision 0 (Issue: The Use of a Demonstration Program as Confirmation of Integrity for Continued Storage of High Burnup Fuel Beyond 20 Years) (ML13056A516)). The NRC is allowing the continued use of high-burnup fuel, even though the NRC recognizes that further studies are needed to determine whether high-burnup fuel can be safely moved from a spent fuel pool to dry cask storage. Ex. 2. While the DGEIS lists some of these references, it never discusses whether high-burnup fuel creates more serious problems than normal spent fuel. The attached Exhibit 1 provides new information that the NRC must now address in a supplemental DGEIS.

In particular, the supplemental DGEIS must, at a minimum, provide a required bounding calculation that considers the consequences of high-burnup fuel. Instead, the DGEIS relies on The Technical Study of Spent Nuclear Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, NUREG–1738 (2001) (ML010430066). The DGEIS cites that 2001 study as the primary reference for its conclusion regarding spent fuel pool fires during a period of 60 years beyond the operating life of the reactor. DGEIS at xxix, F-14. But that 60 year period could include more than 20 years of high-burnup fuel storage in the spent fuel pool. Recent analyses, such as Exhibit 1, make clear that by that time there could be significant deterioration of fuel cladding, making movement of high-burnup fuel from the pool to dry casks problematic. The DGEIS does not take the NEPA-required "hard look" at this or any of the other special problems created by high-burnup fuel.

The DGEIS's conclusion that spent fuel pool storage is environmentally safe also ignores known facts about high-burnup fuel. For example, the DGEIS indicates that the NRC "is aware of concerns regarding potential detrimental effects of hydride reorientation on cladding behavior (e.g., reduced ductility). Reduced ductility, which makes the cladding more brittle, increases the difficulty of keeping spent fuel assemblies intact during handling and transportation." *Id*. at B-13. But the DGEIS contains no discussion of how this recognized "difficulty" affects transferring this fuel from spent fuel pools to dry cask storage, and contains only a cursory discussion of the problems with moving high-burnup fuel from one dry cask to another. *Id*.

Finally, because of the special problems created by high-burnup fuel and the uncertainties inherent in its current use, the DGEIS fails to consider the alternative of prohibiting the further generation of high-burnup fuel until the unresolved safety problems with its use have been addressed. That alternative would have the advantage of allowing the movement of spent fuel from spent fuel pools to dry casks sooner, allowing for a reduction of the crowding of the spent fuel pools and reducing both the risk and the consequences of a severe spent fuel pool accident.

**II. New and Significant Information on Spent Fuel Pool Accident Consequences**

The DGEIS asserts that earlier studies of spent fuel pool accident consequences, like NUREG-1738, were too conservative. DGEIS at F-4 to F-5. New and significant information, including recent analyses of the Fukushima accident, makes clear that those studies in fact underestimated the real potential adverse impacts of a severe spent fuel pool accident.

The NRC has stated that a central part of the input for the DGEIS is the Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor (October 2013) ("Consequences Study") and the related COMSECY-13-0300. *See e.g.*, Ex. 3 (NRC Slides for 8-22-13 Meeting with Commissioners re: Tier 3 Issues, Slide 4 ("Schedules are aligned to improve the public's ability to understand the relationships between the Tier 3 issue, the SFPS, ongoing Waste Confidence activities, and related policy issues.")). Both of those documents address whether expedited transfer of spent fuel to dry cask storage would be preferable to using spent fuel pools for 60 years after reactor operation ceases. Central to those analyses, and to the accident analysis in Appendix F of the DGEIS, is the MELCOR Accident Consequence Code Systems-2 ("MACCS2") code.

The New York Attorney General's Office has submitted comments explaining in detail some of the flaws underlying the inputs used by the MACCS2 code. *See* International Safety Research, Inc., Review of Waste Confidence Generic Environmental Impact Statement, ISR Report 13014-01-02, 20 December 2013 ("ISR Report"). Since the time of the ISR Report, additional information makes clear that the post-accident situation is much longer and the cleanup following the accident is much more difficult than is assumed in the DGEIS.

In particular, the real world experience of the Fukushima accident is far different than what the DGEIS assumes, in terms of (1) the problems created by the need to decontaminate a large area; (2) the time and money required for cleanup; and (3) the lost economic revenue when a large area is rendered unusable for a much longer time than was assumed in the DGEIS. *See* Ex. 4 (David McNeil, *Squelching Efforts to Measure Fukushima Meltdown* (NY Times March 16, 2014)) (explaining how the actual damage caused by Fukushima may be much greater than reported by Japan and that just removal of contaminated dirt—not its ultimate disposal—will cost at least $50 billion); Ex. 5 (*Fukushima operator restarts water decontamination system* (AFP March 24, 2014)) ("The embattled firm [TEPCO] said two of three lines that clean the toxic water were running again as of Monday afternoon. A third line remained offline while workers tried to fix a filter defect which had prevented proper decontamination. . . . TEPCO is struggling to handle a huge—and growing—volume of contaminated water at the tsunami-damaged plant. There are about 436,000 cubic metres of contaminated water stored at the site in

about 1,200 purpose-built tanks."); Ex. 6 (*Contaminated water still troubles Fukushima* (Press TV March 11, 2014)) ("The radioactive water at Japan's crippled nuclear power plant remains the biggest problem, hampering the cleanup process three years after the disaster, officials say. On Monday, officials at Japan's crippled Fukushima nuclear power plant said the contaminated water accumulated at the facility was hampering the cleanup process."); Ex. 7 (*Fukushima water decontamination might be suspended indefinitely* (Rt.com March 20, 2014)); *see also* D. Lochbaum et. al., *Fukushima—The Story of a Nuclear Disaster* (New Press 2014).

This recently disclosed information about Fukushima contrasts sharply with the DGEIS. For instance, the DGEIS assumes that the total economic cost of a full release of radiation from a spent fuel pool would be around $55 billion. DGEIS at F-4. As noted above, one recent analysis of Fukushima has estimated that it would cost that much money just to remove the contaminated soil, which is only one of many costly steps in the process of radiological decontamination. *See* Ex. 4. This requires the NRC to issue a supplemental DGEIS that incorporates this information, which is more in line with a previous NRC study that noted a high estimate for a full pool release as an economic cost of $566 billion, not including health effects and 143,000 latent fatalities. Ex. 8 (Travis et al., *A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants*, NUREG/CR-6451 (1997)) at 4-2. The DGEIS notes that NUREG/CR-6451 provides "reasonable bounding estimates for offsite consequences for the most severe accidents," DGEIS at B-11—a conclusion that is reinforced by recent analyses of Fukushima—but then fails to apply those estimates in its offsite consequences analysis. In light of the recent studies and analyses of Fukushima, the NRC must issue a supplemental DGEIS addressing these analyses and addressing previous studies borne out by the new and significant information about Fukushima. *See id.*; Ex. 9 (Alvarez et al., *Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States* (Science and Global Security, 11:1–51, 2003)); *see also* U.S. Senate Committee on Environment and Public Works, Subcommittee on Clean Air and Nuclear Safety, "Oversight Hearing: NRC's Implementation of the Fukushima Near-Term Task Force Recommendations and other Actions to Enhance and Maintain Nuclear Safety" (SD-406) (Jan. 30, 2014) (Chair Macfarlane at 1:28:10: "There was no evidence that a Fukushima-type accident would have been completely avoided in the US. . . . We did not, prior to the Fukushima accident, expect or analyze for more than one reactor at a site to have an accident. That was not planned for. . . . And the operating experience that we've gained during the Fukushima accident is significant."; Chair MacFarlane at 1:51:54: "Passive systems are certainly better than active systems—systems that have to be activated. So those passive systems are certainly an improvement.").

In addition to these recent analyses of the Fukushima accident, new and significant information from the NRC Staff also calls into question the DGEIS's underlying assumption that spent fuel pool accidents can be analyzed generically. In particular, the NRC Staff—after the close of the comment period for the DGEIS—issued a draft guidance document that specifically recommends requiring a "site-specific analysis" of spent fuel pool accidents before the NRC can exempt decommissioned plants from emergency planning requirements. Ex. 10 (Interim Staff Guidance on Emergency Planning Exemption Requests For Decommissioning Nuclear Power Plants, NSIR/DPR-ISG-02 (January 10, 2014)) at 6. A supplemental DGEIS is required to provide the public with an opportunity to comment on why the NRC would allow a generic analysis in the DGEIS when site-specific analysis is required elsewhere.

## III. New and Significant Information on the Failure of Institutional Controls

When the Commission abandoned the attempt to predict when, if ever, a permanent waste repository would come into existence, one Commissioner observed that "this is a particularly difficult time to be in the prediction business." Comments of Commissioner Svinicki on SECY - 09-0090 Final Update of the Commission's Waste Confidence Decision (Sept. 24, 2009). Despite this warning, the current DGEIS includes a number of assumptions about what will happen decades, centuries, or even millennia into the future. *See, e.g.*, DGEIS at 4-76 to 4-79 & B-15 to B-17. One of those predictions—that institutional controls will provide vigorous regulation and enforcement of safety measures—cannot withstand scrutiny in light of recent events. In particular, the most recent examples of the breakdown in safety involving nuclear wastes occurred at the Hanford Reservation in Hanford, Washington and at the Waste Isolation Pilot Project ("WIPP") in New Mexico. These recent events—discussed in detail below and not considered in the DGEIS—demonstrate why the DGEIS should not assume that NRC regulations will avoid significant problems in the future and ensure that any problems are addressed appropriately. If there is one over-arching lesson from Fukushima, it is that things can go terribly wrong. The following events are further proof of that truth.

First, the Hanford Reservation in Hanford, Washington, despite extensive oversight and numerous measures to avoid releases of radioactive waste, continues to leak radioactive materials. On March 21, 2014—well after the close of the comment period for the DGEIS—the Washington Department of Ecology issued an Administrative Order in Docket 10156 against the United States Department of Energy because of serious leaks of radioactive materials from storage. Ex. 11. The Administrative Order found the following violations:

> **Violation 1 - Failure to stop the flow of hazardous waste into secondary containment.**
> *40 CFR 265.196(a) requires the owner or operator of the tank to immediately stop the flow of hazardous waste into the secondary containment system*.
> As of the date of this Order, USDOE and WRPS have not stopped the flow of waste into the secondary containment of 241-AY-102.
>
> **Violation 2 - Failure to inspect the tank to determine the cause of the release.**
> *40 CFR 265.196(a) requires the owner or operator of the tank to inspect the tank to determine the cause of the release.*
> As of the date of this Order, USDOE and WRPS have not inspected the tank to determine the cause of the release. USDOE states in the revised Pumping Plan that Tank 241-AY-102 will have to be emptied to determine the cause of the release. USDOE has not emptied the tank and has submitted a plan according to which waste removal will not be authorized, nor a removal schedule determined, before March 4, 2016. The revised plan does not demonstrate that an initial pumping date sometime after March 4, 2016 is the earliest practicable time to begin waste removal.
>
> **Violation 3 - Failure to remove, at the earliest practicable time, as much of the waste as is necessary to prevent further release of hazardous waste to the environment and to allow inspection and repair of the tank to be performed.**

> *Where the release is from the tank system, as it is here, 40 CFR 265.196(b) provides that "the owner or operator must, within 24 hours after detection of the leak or, if the owner or operator demonstrates that that is not possible, at the earliest practicable time remove as much of the waste as is necessary to prevent further release of hazardous waste to the environment and to allow inspection and repair of the tank system to be performed."*
> As of the date of this Order, USDOE and WRPS have failed to remove, or take any actions to begin removing, as much of the waste as is necessary to prevent further release to the environment and to allow for inspection and repair of the tank system to be performed. USDOE states in its revised Pumping Plan that removing the contents of the tank will not be authorized before March 4, 2016. USDOE has not demonstrated that March 4, 2016, or later would be the "earliest practicable time" to begin removing the waste.
>
> **Violation 4 - Failure to remove all released materials from the secondary containment system within 24 hours or in as timely a manner as is possible to prevent harm to human health and the environment.**
> *40 CFR 40 CFR 265.196(b)(2) requires that, if the release was to a secondary containment system, all released materials must be removed within 24 hours or in as timely a manner as is possible to prevent harm to human health and the environment.*
> As of the date of this Order, USDOE and WRPS have failed to remove any of the released materials from the secondary containment. The revised plan indicates that the released materials will be removed only after waste is removed from the primary tank.

Ex. 11 at 6-7 (emphasis in original).

The DGEIS does not address the current failures at Hanford or explain how future storage of nuclear waste will be more successful than it is today. The recent events at Hanford provide new and significant information that undermines the DGEIS's assumption that the NRC's regulation of spent fuel storage will avoid serious failures to contain radiation in the future. A supplemental DGEIS must address the recent Administrative Order, as well as the context of past failures to contain high level waste at Hanford. *See, e.g.*, Ex. 12 (R. Alvarez, *Reducing the Risks of High-Level Radioactive Wastes at Hanford* (Science and Global Security 2005) at 13:43–86).

Second, there is new and significant information about a February 2014 release of radiation from the WIPP facility in New Mexico:

> According to the U.S. Department of Energy (DOE), at about 11:30 p.m. (MT) on February 14, 2014, airborne radiation was detected by an underground air monitor at the DOE's Waste Isolation Pilot Plant (WIPP). The source of the radiation is believed to be one or more radioactive waste containers that were breached by an undetermined event that occurred in the underground repository. However, an investigation in the underground is necessary and currently underway to determine the true cause of the release.

Ex. 13 (EPA, *Radiological Event at the WIPP*, http://www.epa.gov/rpdweb00/news/wipp-news.html#wippradevent); *see also* Exs. 14-18 (attachments to Exhibit 13); Ex. 19 (Jeff Tollefson, *Radiation Levels Fall after Nuclear Waste Leak in New Mexico* (Feb. 26, 2014), http://www.scientificamerican.com/article/radiation-levels-fall-after-nuclear-waste-leak-in-new-mexico). This currently unexplained radiation leak underscores the inherent uncertainties in handling high level nuclear wastes—uncertainties that are ignored in the DGEIS.

The WIPP radiation leak occurred 9 days after another accident at the WIPP involving a fire inside the mine. Although it appears radiation was not released during that fire, a DOE investigation of this event found "the root cause of this accident to be the failure of Nuclear Waste Partnership LLC (NWP) and the previous management and operations (M&O) contractor to adequately recognize and mitigate the hazard regarding an underground fire. This includes recognition and removal of the buildup of combustibles through inspections and periodic preventative maintenance (e.g., cleaning), and the decision to deactivate the automatic onboard fire suppression system." Ex. 20 (Accident Investigation Report, Underground Salt Haul Truck Fire at the Waste Isolation Pilot Plant February 5, 2014 (March 2014)) at ES-3. The Accident Investigation Report includes a long list of deficiencies in the operation of this disposal facility and recommends substantial corrective actions. *Id*. at 92-97. The Report also notes that these problems arose despite the clearly stated mission of the Carlsbad Field Office of DOE to store radioactive waste safely through protection measures "put into operation at all levels (site, facility, task, and activity) by requiring and routinely verifying that work is conducted following" all applicable protocols. *Id.* at 64. NRC regulations contain similar protocols and statements, and the recent incidents at the WIPP make clear that where nuclear wastes are concerned, even the best intentions do not prevent serious accidents.

The recent Hanford and WIPP incidents are particularly relevant to the DGEIS in light of the NRC's Office of Inspector General's conclusion that the "NRC's approach for oversight of licensees' management of active component aging is not focused or coordinated" and lacks "mechanisms for systematic and continual monitoring, collecting, and trending of age-related data for active components." Ex. 21 (Audit of NRC's Oversight of Active Component Aging, OIG-14-A-02 (Oct. 28, 2013)) at ii. That same office had previously found deficiencies in NRC's follow-up to assure that licensees fulfill commitments they have made to assure adequate protection of the public health and safety. Ex. 22 (Audit of NRC's Management of Licensee Commitments OIG-A-17 (Sept. 19, 2011)). These reports make clear that NRC regulation can be subject to the same kinds of institutional deficiencies that led to the incidents at the WIPP.

The fact that the NRC and DOE have had problems managing nuclear waste is not a reflection of failures of these agencies—to the contrary, it shows that even with competent and committed staff and leadership, things can go wrong. That is the history of nuclear waste storage, and it is what the NRC must assume going forward, particularly when attempting—as the DGEIS does—to forecast decades, centuries, or even millennia into the future. Or as it is written on the face of the National Archives, "What is past, is prologue." Given this history, highlighted by the new and significant information on the Hanford and WIPP incidents, the DGEIS should not assume that future oversight and future technical developments will eliminate future problems. When it comes to handling nuclear waste, history demonstrates that optimistic assumptions about containment—such as those in the DGEIS—do not become realities.

For the above reasons, the State of Vermont, the State of Connecticut, and the Commonwealth of Massachusetts respectfully request that the NRC Staff prepare a supplemental waste confidence DGEIS in light of recent information and events. Thank you for your consideration of this request.

Sincerely,

Susanne R. Young
Deputy Attorney General
Vermont Attorney General's Office
109 State Street
Montpelier, Vermont
05609-1001
syoung@atg.state.vt.us

Darren M. Springer
Deputy Commissioner
Anthony Z. Roisman
Of Counsel
Vermont Department of Public Service
112 State Street
Montpelier, Vermont 05602
Darren.Springer@state.vt.us
aroisman@nationallegalscholars.com

Robert Snook
Assistant Attorney General
State of Connecticut
Office of the Attorney General
55 Elm Street, P.O. Box 120
Hartford, Connecticut 06106
robert.snook@ct.gov

John Beling
Assistant Attorney General
Commonwealth of Massachusetts
Office of the Attorney General
One Ashburton Place
Boston, MA 02108
john.beling@state.ma.us

# EXHIBIT 1

# The Storage and Disposal Challenges of High Burnup Spent Power Reactor Fuel

**By**

**Robert Alvarez, an Institute for Policy Studies senior scholar**
**January 3, 2014**

## Introduction

Beginning in the 1990s, the U.S. Nuclear Regulatory Commission (NRC) effectively permitted U.S. reactor operators to double the amount of time nuclear fuel can be irradiated in a reactor — by approving an increase in the percentage of uranium-235, the key fissionable material that generates energy. In doing so, NRC bowed to the wishes of nuclear reactor operators, motivated more by economics than spent nuclear fuel storage and disposal.

Known as increased "burnup" this practice is described in terms of the amount of electricity in gigawatts (GW) produced per day with a ton of uranium.

Reactor fuel burnups have gradually increased on the average to ~50 GWd/t for pressurized reactors (PWR) and 43GWd/T for boiling water reactors (BWR). [1] Projected burnups are estimated to increase. (See Figure 1) The current maximum peak burnup limit is 62MWd/t. Reactor operators would like to increase burnups to 75GWd/t..[2] As of 2008, the NRC allowed reactors using uranium fuel to operate at the highest burnup rates of any country in the world.[3]

[1] E. Supko, Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling, Revision 1, Electric Power Research Institute, August 2012.

[2] V. Jain, G. Cragnolino and L. Howard, A review Report on High Burnup Spent Nuclear Fuel Disposal Issues, Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas, CNWRA 2004-08, September 2004, p.xv.

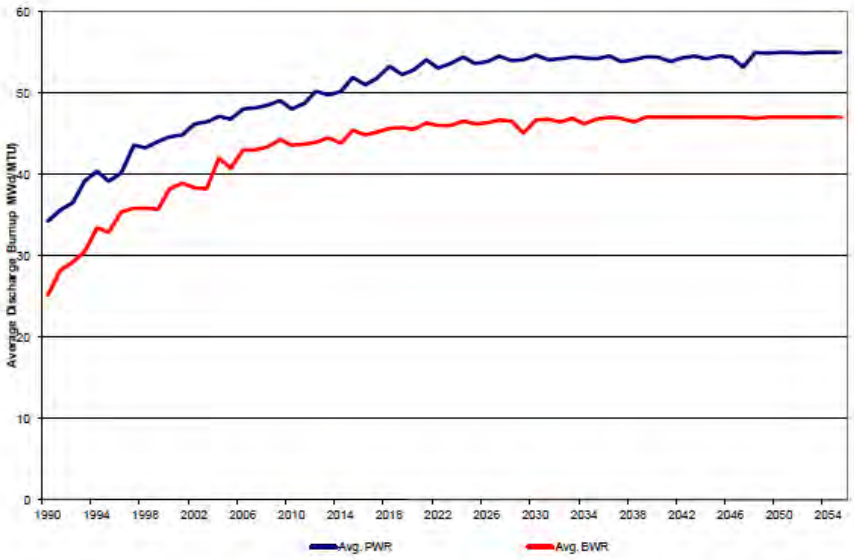
3 Erik Kolstad, Nuclear Fuel Behaviour in Operational Conditions and Reliability, Prepared for IPG meeting-Workshop on Fuel Behaviour, Argonne National Laboratory, September 2008, p. 10

As uranium fuel is irradiated in a reactor core, radioactive elements are created when the atoms of uranium-235 and other heavy isotopes are split (fission) as well as by absorption (activation) of neutrons in the atoms of many other isotopes. The fuel is enriched above its naturally-occurring fraction of 0.7 percent of U-235 to as much as 4.8 percent so it can serve as the primary isotope needed for fission and thus, the generation of energy.

## Inadequate Technical Basis

While the move to high burnup in U.S. power reactors has improved the nuclear power sales, it remains a significant impediment to the safe storage and disposal of spent nuclear fuel. For more than a decade the problems and concerns associated with high burnup spent nuclear fuel have increased, while the resolution of these problems remains illusive. For instance:

**Figure 1. Historical and Projected Average BWR and PWR Discharge Burnups (Source: Supko/EPRI 2012)**




- In 2000, several years after granting increased burnups for U.S. power reactors the U.S. Nuclear Regulatory Commission admitted, "There is limited data to show that the cladding of spent fuel with burnups greater than 45,000 MWd/MTU will remain undamaged during the licensing period." [4]

4 U.S. Nuclear Regulatory Commission, Standard Review Plan for Spent Fuel Dry Storage Facilities, Final Report NUREG-1567, March 2000. P. 6-15. http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1567/sr1567.pdf

- In 2003 the Electric Power Research Institute concluded: "For the most part, the current licensing basis for dry storage of spent fuel is largely based on fuel examinations and dry storage performance demonstrations performed in the 1980s and 1990s. Spent fuel used in the dry storage performance demonstrations had discharge burnups of ~36 GWd/MTU, or less."[5]

- In 2010, researchers at Oak Ridge National Laboratory reported to the NRC that "the majority of isotopic assay measurements available to date involve spent fuel with burnups of less than 40 GWd/MTU and initial enrichments below 4 wt % 235U, limiting the ability to validate computer code predictions and accurately quantify the uncertainties of isotopic analyses for modern fuels in the high burnup domain."[6]

- That same year the Nuclear Waste Technical Review Board reported: "Only limited references were found on the inspection and characterization of fuel in dry storage, and they all were performed on low-burnup fuel after 15 years or less of dry storage. Insufficient information is available yet on high-burnup fuels to allow reliable predictions of degradation processes during extended dry storage, and no information was found on inspections conducted on high-burnup fuels to confirm the predictions that have been made."[7]

- In 2012, EPRI reported that "R&D work will continue especially in concert with introduction of new cladding materials" [and] "R&D work will continue especially in concert with introduction of new cladding materials...[and a] Key question: Given what we learned, how does that knowledge support existing — or coming up with new — regulatory guidance?" [8]

- In 2012, the official publication of the National Academy of Engineering of the National Academy of Sciences raised similar concerns about the viability of high-burnup fuel by noting "the technical basis for the spent fuel currently being discharged (high utilization, burnup fuels) is not well established… the NRC has not yet granted a license for the transport of the higher burnup fuels that are now commonly discharged from reactors. In addition, spent fuel that may have degraded after extended storage may present new obstacles to safe transport."[9]

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[5] Electric Power Institute, Dry Storage Demonstration for High-Burnup Spent Nuclear Fuel Feasibility Study, September 2003, p.5-1.

[6] G. Ilas and I.C. Gauld, Analysis of Experimental Data for High-Burnup PWR Spent Fuel Isotopic Validation—Vandellós II Reactor, ORNL/TM-2009/32, p. 1. http://info.ornl.gov/sites/publications/files/Pub22621.pdf

[7] United States Nuclear Waste Technical Review Board, *Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel, December 10, 201*0.

[8] Albert Machiels, Electric Power Research Institute, High-Burnup – 10 Years Later, Used Fuel and HLW Management Technical Advisory Committee Washington, DC September 13, 2012

9 National Academy of Engineering, Managing Nuclear Waste, Summer 2012, pp 21, 31. http://www.nae.edu/File.aspx?id=60739

## Impacts

EPRI pointed out in 2005 that: "*Failure to resolve, in a timely manner, regulatory issues associated with interim dry storage and transportation of high-burnup spent fuel would result in severe economic penalties and in operational limitations to nuclear plant operators.* [Emphasis added.]"[10]

Analysts with Advanced Nuclear technology International concluded in 2012 that "the inability to send fuel for reprocessing or to a permanent storage site has caused a spent fuel assembly log-jam in the spent fuel pools and effectively eliminated this high burnup incentive…the significantly increased time required for high burnup fuel to decrease its decay heat in a spent fuel pool before it can be loaded into an intermediate dry storage cask and the unknown schedule for shipping the fuel from the dry cask to a permanent storage site prevents a reliable estimate for the capacity and cost required for the intermediate wet and dry storage facilities."[11]

There remain several issues of concern that impact the storage and disposal of high-burnup spent nuclear fuel. Several hundred pellets made of slightly enriched ceramic uranium dioxide (U02) are stacked in zirconium metal alloy tubes and sealed at both ends. The gap between the rods and pellets of approximately 152 micrometers is filled with helium to a pressure of 10 bar or 145 pounds per square inch. Thickness of the rod cladding is between 0.04-0.8 mm (0.00157 to 0.00314 inches)[12] — 15 to 30 times less than a computer disc (CD/DVD) [13] and slightly thicker than extra heavy duty aluminium foil used in kitchens. [14]

With higher burn up, nuclear fuel rods undergo several potentially risky changes that include:

- Increasing oxidation, corrosion and hydriding of the fuel cladding. Oxidation reduces cladding thickness, while hydrogen (H3) absorption of the cladding to form a hydrogen-based rust of the zirconium metal from the gas pressure inside the rod can cause the cladding to become brittle and fail;[15]
- Higher internal rod gas pressure between the pellets and the inner wall of the cladding leading to higher fission gas release. Pressure increases are typically two to three times greater.[16]

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[10] Electric Power research Institute, Application of Critical Strain Energy Density to Predicting High-Burnup Fuel Rod Failure, September 2005, P.vi.

[11] Peter Rudling, Charles Patterson, Ron Adamson, Friedrich Garzarolli, Alfred Strasser, Tony Turnbull, Special Topic Report: High Burnup Fuel Design Issues and Consequences, Advanced Nuclear Technology International, IZNA 12,p.5-1. https://www.antinternational.com/fileadmin/Products_and_handbooks/sample/IZNA12/IZNA12_STR_HiBu_sample.pdf

12 U.S. Department of Energy, Argonne National Laboratory, Nuclear Fuel, April 2011.

13 Graham Sharpless, CD and DVD Disc Manufacturing, Deluxe Global Media Services Ltd., July 2003.p 3. http://cddvdreplications.com/help/technology/downloads/tech_docs/replication.pdf

14 Desmond Fraser, Electromagnetic Engineering and Aluminum Foil, Rheintech Laboratories, Inc., August 3, 2011. http://www.rheintech.com/blog/archives/884

15 U.S. Nuclear Regulatory Commission, Rulemaking Issue, Notation Vote, Memorandum from: R.W. Borchardt, Executive Director for Operations, Subject: Proposed Rulemaking – 10CFR 50.46c Emergency Core Cooling System Performance During Loss-of-Coolant Accidents (RIN 3150-AH42), SECY-12-0034, March 1, 2012, p. 2. http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2012/2012-0034scy.pdf

16U.S. Nuclear regulatory Commission, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning

- During a power release at high burnup cladding can deform and fail.[17]
- Elongation or thinning of the cladding from increased internal fission gas pressure;[18]
- Structural damage and failure of the cladding caused by hoop (circumferential) stress; [19]
- Increased debris in the reactor vessel, damaging and rupturing fuel rods;[20]
- Cladding wear and failure from prolonged rubbing of fuel rods against grids that hold them in the assembly as the reactor operates (grid to rod fretting).[21]
- Oxidation of irradiated fuel pellets during extended storage.[22]
- A significant increase in radioactivity and decay heat in the spent fuel. [23]
- A potentially larger number of damaged spent fuel assemblies stored in pools[24]
- Upgraded pool storage with respect to heat removal and pool cleaning.[25]
- Requiring as much as 150 years of surface storage before final disposal. [26]
- Increased costs for disposal due to decay heat.[27]
- Potential repository criticality [28]
- Increased radiation doses following geologic disposal[29]
- Swelling and closure of the pellet-cladding gap- increasing cladding stresses, creep and stress corrosion cracking of cladding in extended storage. [30]
- Embrittlement of cladding due to decreases in fuel temperatures during extended storage.[31]

There is growing evidence that as a result of higher burn-ups nuclear fuel cladding cannot be relied upon as a primary barrier to prevent the escape of radioactivity, especially during dry storage. This has not been lost on the nuclear industry and staff of the NRC for several years

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Nuclear Power Plants, October 2000, P. 45. http://pbadupws.nrc.gov/docs/ML0104/ML010430066.pdf

[17] Stafano Caruso, c*haracterisation of high-burnup LWR fuel rods through gamma tomography*, École Polytechnique Fédérale De Lausanne, April 2007.

18 Op cit ref. 12.

19 Ibid

20 International Atomic Energy Agency, Impact of High-Burnup Uranium Oxide and Mixed Uranium – Plutonium Oxide Water Reactor Fuel on Spent Fuel Management, IAEA Nuclear Energy Series, No.. NF-T-3.8, June 2011. P. 39. http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1490_web.pdf

21 Ibid.

[22] Op Cit Ref. 7.

23 Op. cit ref. 16.

24 Ibid p. 51.

25 Ibid. p.1.

26 Zhiwen Xu, Mujid S. Kazimi and Michael Driscoll, Impact of High Burnup on PWR Spent Fuel Characteristics, Nuclear Science and Engineering, 151, 261-273 (2005), http://ocw.internet-institute.eu/courses/nuclear-engineering/22-251-systems-analysis-of-the-nuclear-fuel-cycle-fall-2005/readings/impact.pdf

[27] Ibid.

[28] Zhen Xu, Designing Strategies for Optimizing High Burnup in Pressurized Reactors, Massachusetts Institute of Technology, Department of Nuclear Engineering, January 2003.

[29] Sitakanta Mohanty, Lynn Tipton, Razvan Nes, and David Pickett, High-Burnup of Spent Nuclear Fuel and Its Implications for Disposal Performance Assessments, Symposium on the Scientific Basis for Nuclear Waste Management XXXVI at the 2012 Materials Research Society Fall Meeting, Boston, Massachusetts, USA, November 25─30, 2012

[30] op cit. Ref 7.

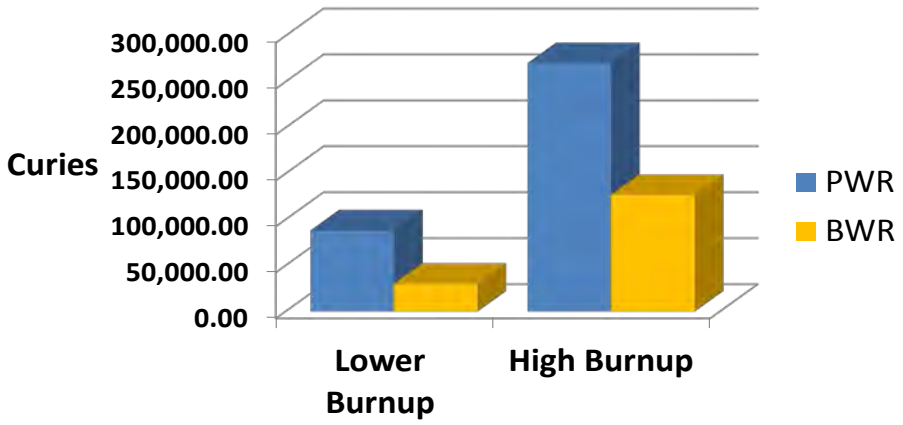
[31] Ibid.

now. Damage in the form of pinhole leaks, and small cracks that could lead to breaching of fuel cladding is "not explicitly defined in [NRC] Regulations, staff guidance or standards."[32]

## Source Term and Decay Heat

Given these uncertainties the U.S. Department of Energy (DOE) and the NRC have provided general estimates of the radionuclide content of spent nuclear fuel based on current and previous burnup assumptions. According to DOE the estimated average long-lived radioactivity for a typical PWR and BWR assembly having lower burnup at the time of geological disposal are 88,173.69 curies and 30,181.63 curies respectively. [33] For current burnups the NRC estimates that the post discharge radioactive inventory of spent fuel for a typical PWR and BWR assemblies are 270,348.26 curies and 127,056.67 curies respectively (See Figure 2). [34] Approximately 40 percent of the total estimated radioactivity for lower and high burnup is Cs-137.

**Figure 2. estimated radioactivity in a U.S. spent nuclear fuel assembly**




**Sources: DOE EIS-0250, Appendix A, http://energy.gov/sites/prod/files/EIS-0250-FEIS-01-2002_0.pdf**
**NRC http://pbadupws.nrc.gov/docs/ML0907/ML090770390.pdf**

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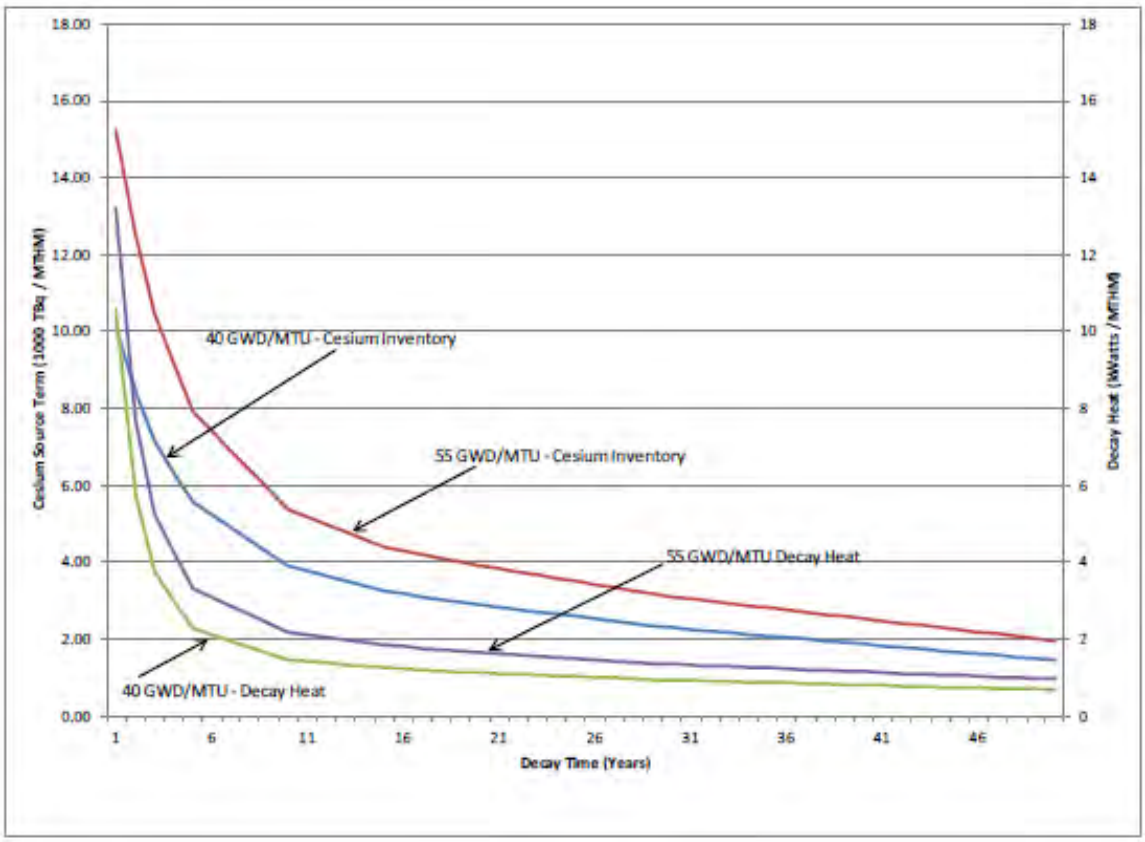
[32] RE Einziger et al., Damage in Spent Nuclear Fuel Defined by Properties and Requirements, U.S. Nuclear Regulatory Commission, Spent Fuel Project Office, June 2006. http://pbadupws.nrc.gov/docs/ML0608/ML060860476.pdf

[33] U.S. Department of Energy, Final Environmental Impact Statement, for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada, 2002, Appendix A, Tables A-7, A-8, A-9, A-10, (PWR/ Burn up = 41,200 MWd/MTHM, enrichment = 3.75 percent, decay time = 23 years. BWR/ Burn up = 36,600 MWd/MTHM, enrichment = 3.03 percent, decay time = 23 years.)

[34] U.S. Nuclear Regulatory Commission, Characteristics for the Representative Commercial Spent Fuel Assembly for Preclosure Normal Operations, May 2007, Table 16, p.44-45. http://pbadupws.nrc.gov/docs/ML0907/ML090770390.pdf

This substantial increase in spent nuclear fuel radioactivity has also resulted in a commensurate increase in decay heat. After removal, the spent fuel gives off a significant amount of heat as the radioisotopes decay. After removal, the spent fuel gives off a significant amount of heat as the radioisotopes decay ( see Figures 3 and 4). The offload of a full reactor core at a PWR is estimated to give off about 42,000 BTU/hr (12,310 watts). [35] Within one year the heat output of the spent fuel diminishes by about ten times. The decay heat for a five-year cooled PWR assembly with a discharge exposure of 55 GWd/MTU is approximately 1,500 watts. [36] The decay heat for a five-year cooled BWR assembly with a discharge exposure of 48 GWd/MTU is approximately 480 watts. [37]

## Figure 3. PWR SNF Assembly Decay Heat (right axis) and Cesium Inventory (left axis) as a Function of Burnup and Cooling Time




Source: Supko/EPRI 2012

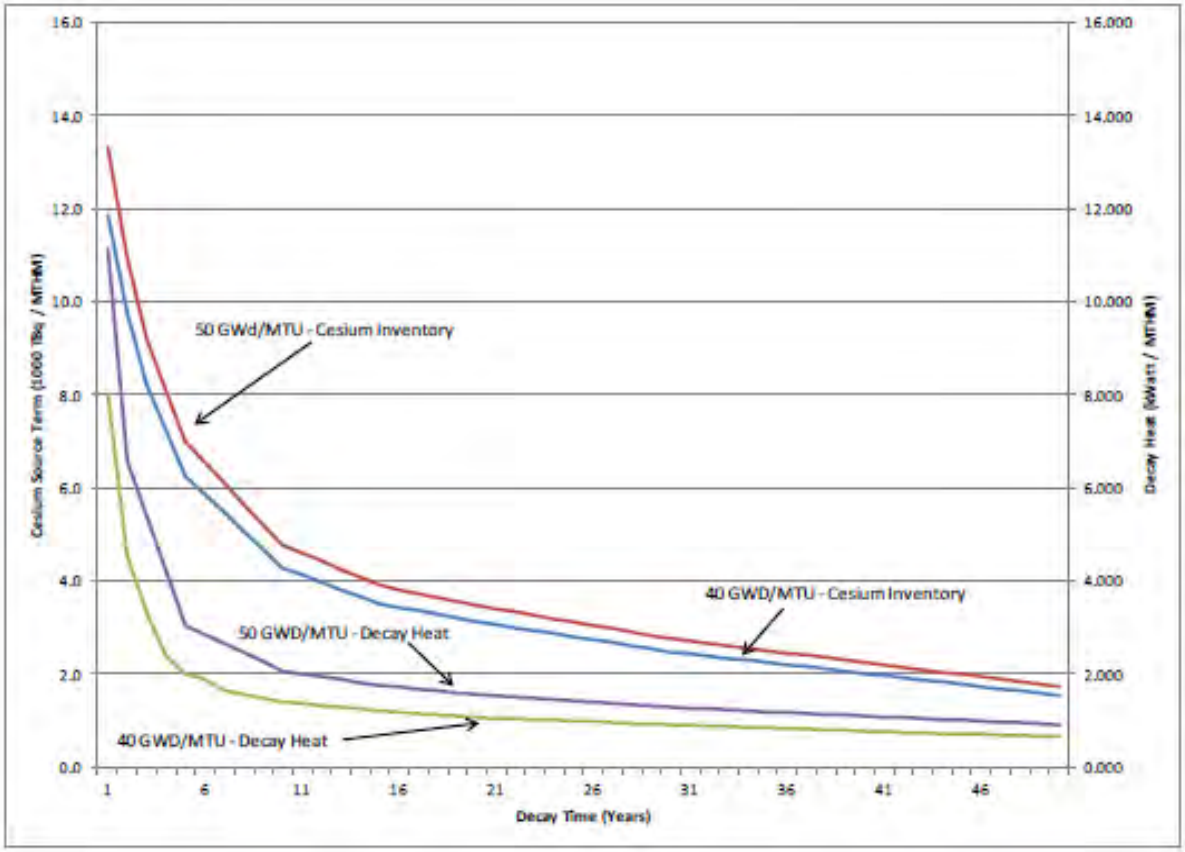
35 U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Safety Regulation Related to Amendment No. 131 to Facility Operating License No. NPF-10 and Amendment 120 to Facility Operating License No. NPF-15, Docket Nos. 50-361 and 50-362, October 1996, P. 6. http://pbadupws.nrc.gov/docs/ML0220/ML022000232.pdf

36 Op Cit Ref.1.

[37] Ibid.

Within one year the heat output of the spent fuel diminishes by about ten times. After 10 years it drops by another factor of ten. By 100 years the decay heat has dropped another five times, but still gives off significant heat.[38] However, the decay heat remains substantially high throughout the operation of the reactors and well after they are closed.

**Figure 4.** ***BWR SNF Assembly Decay Heat (right axis) and Cesium Inventory (left axis) as a Function of Burnup and Cooling Time***




(Source: Supko/E{RI 2012)

38 Op Cit Ref. 4.

Control of decay heat is a key safety factor for spent fuel storage and its final disposal in a geological repository. Storage of spent nuclear fuel in pools requires continuous cooling for an indefinite period to prevent decay heat from igniting the zirconium cladding and releasing large amounts of radioactivity into the environment.

Zirconium cladding of spent fuel is chemically very reactive in the presence of uncontrolled decay heat. According to the National Research Council of the National Academy of Sciences the build up of decay heat in spent fuel in the presence of air and steam:

> " is strongly exothermic — that is, the reaction releases large quantities of heat, which can further raise cladding temperatures… if a supply of oxygen and or steam is available to sustain the reactions.. The result could be a runaway oxidation — referred to as *a zirconium cladding fire* — that proceeds as a burn front (e.g., as seen in a forest fire or fireworks sparkler)...As fuel rod temperatures increase, the gas pressure inside the fuel rod increases and eventually can cause the cladding to balloon out and rupture.[original emphasis] "[39]

The Nuclear Regulatory Commission (NRC) has performed several studies to better understand this problem. In 2001, the NRC concluded:

> "It was not feasible, without numerous constraints, to establish a generic decay heat level (and therefore a decay time) beyond which a zirconium fire is physically impossible."[40]

In terms of geologic disposal, decay heat, over thousands of years, can cause waste containers to corrode, negatively impact the geological stability of the disposal site and enhance the migration of the wastes.[41]

EPRI points out that radiocesium inventories have greatly increased as well as decay heat. It contends that a return to open-rack cooling of SNF would result in a reduction in the potential source term of 43 percent to 53 percent for a PWR and 47 percent to 48 percent for a BWR.

## Wet Storage Issues

The accumulation of high-burnup spent nuclear fuel in pools adds to the growing concern over age and deterioration of spent fuel pool storage systems. A 2011 NRC-sponsored study, concluded," *as nuclear plants age, degradations of spent fuel pools (SFPs), reactor refueling cavities…are occurring at an increasing rate, primarily due to environment-related factors. During the last decade, a number of NPPs have experienced water leakage from the SFPs [spent*

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39 National Research Council, Board on Radioactive Waste Management, Committee on the Safety and Security of Commercial Spent Nuclear Fuel Storage, National Academes Press (2006), p. 38-39. http://www.nap.edu/openbook.php?record_id=11263&page=38 ,http://www.nap.edu/openbook.php?record_id=11263&page=39

40 U.S. Nuclear regulatory Commission, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, October 2000, P. ix. http://pbadupws.nrc.gov/docs/ML0104/ML010430066.pdf

41 R. Wigeland, T.Taiwo, M. Todosow, W. Halsey, J. Gehin, Options Study – Phase II ,Department of Energy, Idaho National Laboratory, INL/EXT-10-20439, September 2010. http://www.inl.gov/technicalpublications/Documents/4781584.pdf

*fuel pools] and reactor refueling cavities.*[42] The authors of this study also indicate that accurate assessment of aging of spent fuel pools is uncertain because, *"it is often hard to assess their in situ condition because of accessibility problems…. Similarly, a portion of the listed concrete structures are either buried or form part of other structures or buildings, or their external surfaces are invisible because they are covered with liners.* [43] .

High-density racks in spent fuel pools at U.S. power plants pose potential criticality safety concerns associated with the deterioration of neutron absorbing panels that allow spent fuel rods to be more closely packed. Since 1983, several incidents have occurred at reactors around the U.S. with these panels in which the neutron-absorbing materials deteriorated, and in some cases, bulged, causing spent fuel assemblies, containing dozens of rods each, to become stuck in submerged storage racks in the pools. This problem could lead to structural failures in the storage racks holding the spent fuel rods in place.

According to the NRC in May 2010:

> The conservatism/margins in spent fuel pool (SFP) criticality analyses have been decreasing…The new rack designs rely heavily on permanently installed neutron absorbers to maintain criticality requirements. *Unfortunately, virtually every permanently installed neutron absorber, for which a history can be established, has exhibited some degradation. Some have lost a significant portion of their neutron absorbing capability. In some cases, the degradation is so extensive that the permanently installed neutron absorber can no longer be credited in the criticality analysis* [emphasis added].[44]

For example, in 2007, South California Edison (SCE) reported to the NRC that Boraflex neutron absorbing panels have deteriorated to the point at the San Onofre Nuclear Generating Station Units 2 and 3 spent nuclear fuel pools where it was doubtful they could be credited to prevent criticality. SCE proposed installing borated stainless steel tube guide inserts, and to add more neutron absorbing boron to the pool water.[45] According to SCE deterioration from erosion, over a period of 15 months, increased the level of particles from disintegrated neutron absorbing panels in the pool water by 134 percent.[46] These particles place an additional strain on pool water cleaning systems.

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[42] U.S. Nuclear regulatory Commission, A summary of Aging Effects and Their Management in Reactor Spent Fuel Pools, Refuelling Cavities, TORI and Safety-Related Concrete Structures, NUREG/CR-7111 (2011). P. vxiii. http://pbadupws.nrc.gov/docs/ML1204/ML12047A184.pdf

[43] Ibid.

44 U.S. NRC, Office of Nuclear Reactor Regulation, On Site Spent Fuel Criticality Analyses, NRR Action Plan, May 21, 2010. http://pbadupws.nrc.gov/docs/ML1015/ML101520463.pdf

45 South California Edison, Letter to the NU.S. Nuclear regulatory Commission, Subject: Docket Nos. 50-361 and 50-362 Amendment Application Numbers 243, Supplement 1 and 227, Supplement 1 Proposed Change Number (PCN)566, Revision 1, Request to Revise Fuel Storage Pool Boron Concentration, San Onofre Nuclear Generating Station Units 2 and 3, June 15, 2007, Enclosure 2,p. 2. http://pbadupws.nrc.gov/docs/ML0717/ML071700097.pdf

46 Ibid.

NRC's response to this problem has been to allow operators to add additional boron to the pool water to compensate for the loss of re-criticality protection from deteriorated neutron absorbing panels. However, boron is implicated in possible deterioration of the reinforced concrete holding the spent fuel pools. Concrete "could be negatively impacted by adverse environments of borated water or where there is the possibility of alkali aggregate material reactivity."[47]

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

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

[47] Op. Cit Ref. 38, p.xiv.

# EXHIBIT 2

**Division of Spent Fuel Storage and Transportation**
**Interim Staff Guidance-24, Revision 0**

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**Issue:** **The Use of a Demonstration Program as Confirmation of Integrity for Continued Storage of High Burnup Fuel Beyond 20 Years**

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," requires that storage of the waste meet various criteria. One criterion requires the spent fuel cladding be protected during storage against degradation that leads to gross ruptures, or not pose operational safety problems with respect to its removal from storage. (see 10 CFR 72.122(h)(1)). Additionally, storage systems must be designed to allow ready retrieval of the waste for further processing or disposal. (see 10 CFR 72.122(l).

This Interim Staff Guidance (ISG) document provides guidance to the staff for reviewing if a demonstration of high burnup fuel (HBF) has the necessary properties to qualify as one method that an applicant might use in license and certificate of compliance (CoC) applications to demonstrate compliance with 10 CFR 72.122(h)(1) and 10 CFR 72.122(l). This guidance is not a regulatory requirement. Alternative approaches may be used to demonstrate safety and compliance, as appropriately justified by an applicant.

**Discussion**:

The experimental confirmatory basis that low burnup fuel (≤45 GWd/MTU) will maintain its integrity in dry cask storage over extended time periods was provided in NUREG/CR-6745 (Ref. 1), "Dry Cask Storage Characterization Project—Phase 1; CASTOR V/21 Cask Opening and Examination" and NUREG/CR-6831 (Ref. 2), "Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage."

A confirmatory basis, which includes information over a similar length of the time available for low burnup fuel, does not exists for HBF (>45 GWd/MTU). Certification and licensing HBF for storage was permitted for an initial 20-year-term using the guidance contained in ISG-11, Rev. 3, (Ref. 3) which was based on short term laboratory tests and analysis that may not be applicable to the storage of HBF beyond 20 years, particularly with the current state of knowledge regarding HBF cladding properties. (Ref. 4)

One concern stated in ISG-11, Rev. 3, was the potential detrimental effects, such as reduced ductility, of hydride reorientation on cladding behavior. Research performed in Japan and the United States indicated that: 1) hydrides could reorient at a significantly lower stress than previously believed, and 2) HBF could exhibit a ductile-to-brittle transition temperature (DBTT) due to the presence of radial hydrides. (Ref. 4) This phenomenon could influence the retrievability of HBF assemblies and result in operational safety problems as HBF cooled. Circumferential zirconium hydrides in the fuel cladding regions would dissolve into the fuel cladding during drying and precipitate (reorient) as radial hydrides as the fuel cladding cooled. Thus, fuel cladding with radial hydrides that is below a DBTT could be too brittle to retrieve on an assembly basis. The maximum temperatures and internal rod pressures, in ISG-11, Rev. 3, were recommended to mitigate hydride reorientation and are applicable to HBF during the initial 20-year storage, as the decay heat of HBF is expected to maintain cladding temperatures above a DBTT (~ 200$^{o}$C).

While there is no evidence to suggest that HBF cannot be safely stored beyond 20 years, data supporting readily retrievable storage of HBF beyond 20 years is not presently available for the time periods used to support retrievability and storage of low burnup fuel. Therefore, confirmatory data or a commitment to obtain data on HBF and taking appropriate steps in an aging management plan (AMP) will provide further information that will be useful in evaluating the retrievability and storage of HBF for more than 20 years.

A demonstration program could provide an acceptable method for an applicant to demonstrate compliance with the cited regulations for storage of HBF for periods of greater than 20 years by:

1. Confirming the expected fuel conditions, based on technical arguments made in ISG-11, Rev. 3, after a substantial storage period (~ 10 years). The behavior of the cladding for the renewal term will depend on its physical condition at the end of the initial 20 year storage period.
2. Providing data for benchmarking, confirming predictive models and updating aging management plans.
3. Justifying the basis for time-limited aging analyses (TLAA). While regulations call for TLAA and an AMP, since an AMP is currently very difficult to implement in a sealed system, data to justify TLAA is imperative.
4. Identifying any aging effects that may be missed through short-term accelerated studies and analyses.

Monitoring of the fuel temperatures and conditions in the cask combined with physical examination of the fuel at periodic intervals should be able to provide confirmation that:

1. The models of the phenomena used for the first 20-year predictions can be used for the TLAA beyond 20 years.
2. The condition of the fuel after 20 years of storage.
3. New degradation mechanisms are not operating.

Extrapolation outside the recorded data carries risk, but that risk can be minimized if the length of the extrapolation is reduced and those extrapolations are updated as the demonstration continues to monitor and measure fuel properties.

**Regulatory Basis:**

- 10 CFR 72.122(h)(1) The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

- 10 CFR 72.122(l) *Retrievability*. Storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related greater than Class C waste for further processing or disposal.

**Applicability**:

This guidance applies to license and CoC applications for the storage of HBF for periods greater than 20 years. This guidance supplements the guidance given in NUREG-1927 on aging management for the interior of the cask. (Ref. 5)

**Technical Review Guidance:**

The applicant may use the results of a completed demonstration or an on-going demonstration if the conditions of the demonstration meet the requirements stated below for the fuels and conditions of storage for which the term is to be renewed. A description of the demonstration, as described in the Safety Analysis Report, shall be incorporated as an enforceable condition that is placed in the CoC. In either case, the demonstration must be in conjunction with an actively updated AMP as an acceptable means for confirming that the canister or cask contents satisfy the applicable regulations. Since limited AMP action can be taken inside a sealed canister, the AMP must ensure that the TLAA is updated with new information as it becomes available, and if the revised TLAA indicates a problem, a plan of action would be developed for mitigation depending on the type and severity of conditions expected.

The following general requirements should be included for a demonstration program for storage of HBF beyond 20 years to be applicable to support a license or certificate application:

1. The maximum burnup of the fuel intended as content in a license application shall be no more than four GWd/MTU greater than the burnup of the fuel used in the demonstration program and shall be of the same cladding type. The demonstration program may have to expand as burnups increase and new fuel claddings are introduced.

2. The demonstration canister will be dried by a widely recognized industry method that results in peak cladding temperatures which bound the peak cladding temperatures requested in the license application. The thermal models used to license the renewal must use the input data from the demonstration to show that the temperatures in the demonstration are bounding.

3. The interior of a helium-filled demonstration canister will be monitored continuously for moisture, hydrogen, oxygen, fission gas, and fuel cladding axial temperature distribution.

4. As a minimum, physical examination of stored rods at periodic intervals to determine cladding creep, fission gas release, hydride reorientation, cladding oxidation and mechanical properties.

5. The demonstration program fuel shall include at least two full fuel assemblies. The assemblies may be reconstituted.

6. Data from the demonstration program must be indicative of a storage duration long enough to justify extrapolation to the total storage time requested but no less than 10 years if the data is to be used to support license extension from 20 – 40 years. The evaluation of the data from the monitoring and examination of individual rods shall be available prior to the end of the currently approved storage period.

An actively updated AMP should be periodically re-evaluated given the current state of knowledge regarding the ability of HBF to meet the regulatory requirements while in dry cask storage. Licensing conditions should require updated AMPs to be submitted to the NRC whenever new data from the demonstration or other short-term tests or modeling indicates potential degradation of the fuel or deviation from the assumptions of the TLAA or AMP. Updates to TLAAs and AMPs will be subject to inspection.

**Recommendations:**

The staff recommends that NUREG-1927 (Ref. 5) be modified to add this guidance in Section 3.4.3 Aging Management Activity.

**Approved:** **[DRAFT]** _____

Mark D. Lombard, Director
Division of Spent Fuel Storage
and Transportation
Office of Nuclear Material Safety
and Safeguards

_____
Date

**References:**

[1] NUREG/CR-6745, "Dry Cask Storage Characterization Project-Phase 1: CASTOR V/21 Cask Opening and Examination." Idaho National Engineering & Environmental Lab, September, 2001.

[2] NUREG/CR-6831, "Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage." Argonne National Laboratory, September 2003.

[3] Interim Staff Guidance - 11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel." U.S. Nuclear Regulatory Commission, November 2003.

[4] M.C. Billone, T.A. Burtseva, and R.E. Einziger, "Ductile-to-Brittle Transition Temperature for High-Burnup Cladding Alloys Exposed to Simulated Drying-Storage Conditions," in the Journal of Nuclear Materials Volume 433, Issues 1–3, pages 431–448, February 2013.

[5] NUREG-1927, "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance Final Report." U.S. Nuclear Regulatory Commission, March 2011.

# EXHIBIT 3

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

*Protecting People and the Environment*

# Japan Lessons Learned Tier 3 Issue:

# Expedited Transfer of Spent Fuel to Dry Cask Storage

*Public Meeting*

August 22, 2013

# Agenda

- Objective & Background
- Regulatory Analysis Process
- Spent Fuel Pool Study Appendix D – Regulatory Analysis and Backfitting Discussion
- Preliminary Outline of Regulatory Analysis for all Spent Fuel Pools

# Presentation Objective

- Inform stakeholders about the staffs activities on the Japan lessons learned Tier 3 activity on expedited transfer of spent fuel
- Discuss the staff's plans for expanding the regulatory analysis contained in the Spent Fuel Pool Study (SFPS) reference plant to make it applicable to all Spent Fuel Pools (SFPs)
- Gather stakeholder feedback for the upcoming Commission paper on this issue

# Background

- Objective of Tier 3 Plan:
  - Determine whether regulatory action needs to be taken to require expedited transfer of spent fuel to dry casks
  - Provides additional regulatory context of the results from the SFPS

- Schedules are aligned to improve the public's ability to understand the relationships between the Tier 3 issue, the SFPS, ongoing Waste Confidence activities, and related policy issues

# Background, cont'd

- Spent Fuel Pool Study initiated in July 2011
- SECY-12-0095 (7/13/2012) established the general plan to address the transfer of spent fuel to dry cask storage
- Related Commission Documents:
  - June 7, 2012 Meeting with ACRS (SRM 7/16/2012)
  - August 7, 2012 Japan Lessons Learned Briefing (SRM 8/24/2012)
  - May 7, 2013 Memorandum to the Commission outlining updated Tier 3 plan

# Tier 3 Plan

- Three phases with Commission papers:
  - Phase 1 – Evaluate whether substantial increase in public health and safety exists (Commission paper by October 2013)
  - Phase 2 – If necessary, perform detailed analysis of costs and benefits (Commission paper by July 2015)
  - Phase 3 – If necessary, consider other factors (criticality, mitigating strategies, solar storms, economic consequences, new regulatory framework, etc.) (Commission paper by July 2017)

# Planned Spent Fuel Storage Regulatory Milestones

| June 2013 | July 2013 | August 2013 | Sept 2013 | Oct 2013 | Nov 2013 |
|---|---|---|---|---|---|
| • 1. Draft report public (June 24) | • ACRS Full Committee (July 9) | | • SFPS - SECY | | |
| • *2. Regulatory Analysis for reference plant* | • *ACRS Sub Committee* • *(July 9)* | • *Public Meeting (Aug. 22)* | | • *1. ACRS Full Committee (Oct .3)* • *2. Phase 1 SECY Public (mid-Oct. )* | |
| • **Commission Review Draft Documents Released (June 24)** | | | • **Publication of Documents (Public Comment Period Opens)** | | • **Public Comment Period Closes (75 days after opening)** |

Legend

Spent Fuel Pool Study

*Tier 3 Expedited Spent Fuel Transfer Plan*

**Waste Confidence**

# What is a Regulatory Analysis?

An analytical tool provided to decision makers which:

- Recommends a preferred alternative from the potential courses of action studied
- Contains estimates of benefits and costs with a conclusion whether the proposed regulatory action is cost beneficial

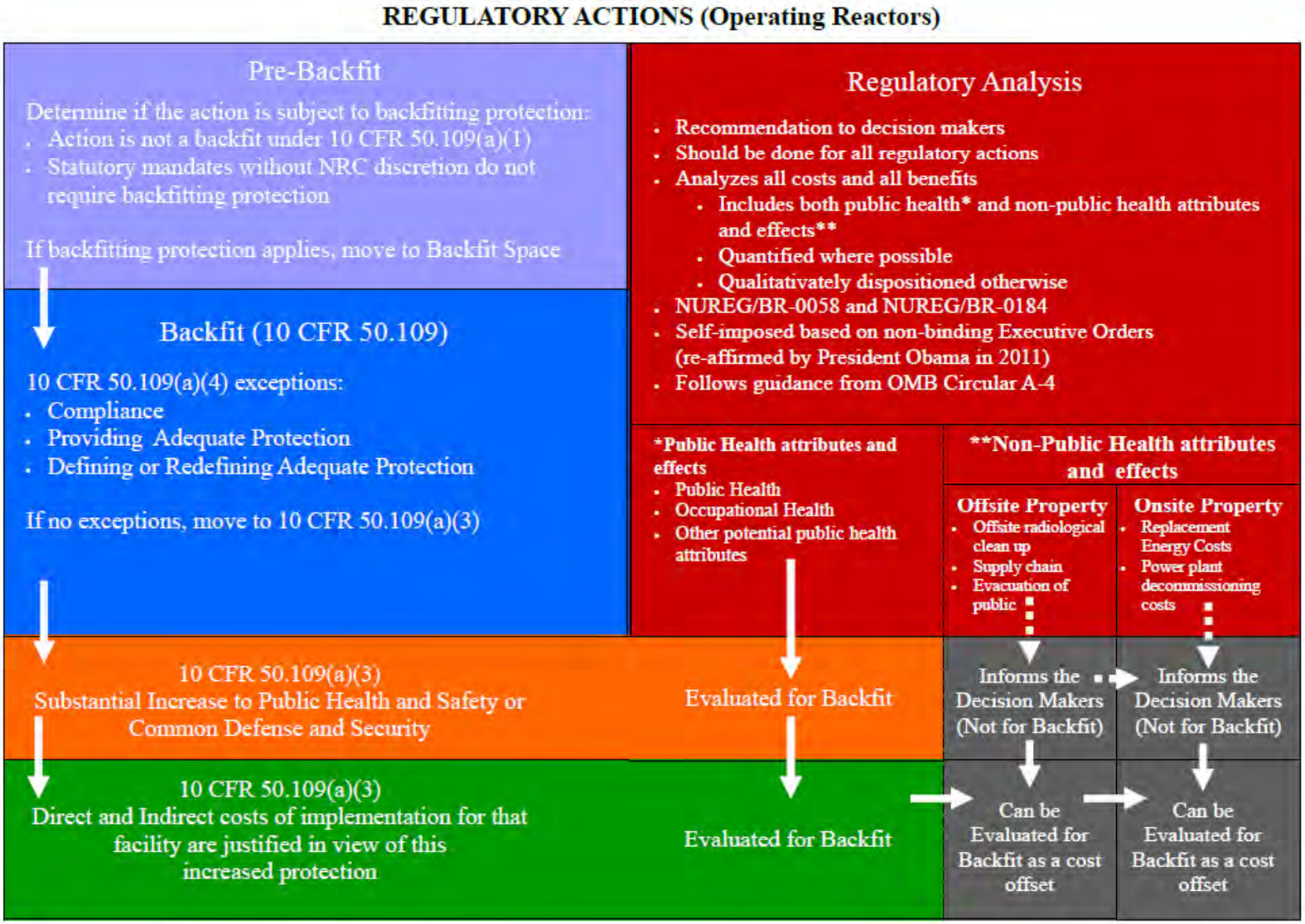
# Elements of a Regulatory Analysis

- Statement of the Problem and Objective
- Identification of Alternatives
- Estimation and Evaluation of Values and Impacts
- Presentation of Results
- Decision Rationale
- Implementation

# Attributes Considered in a Regulatory Analysis

- Public Health (Accident)
- Public Health (Routine)
- Occupational Health (Accident)
- Occupational Health (Routine)
- Offsite Property
- Onsite Property
- Industry Implementation
- Industry Operation
- NRC Implementation
- NRC Operation
- Other Government
- General Population
- Improvements in Knowledge
- Regulatory Efficiency
- Antitrust Considerations
- Safeguards and Security Considerations
- Environmental Considerations
- Other Considerations

# Regulatory Analysis vs. Backfit



# Spent Fuel Pool Study Regulatory Analysis Overview

- The regulatory analysis was performed to provide regulatory context for the Spent Fuel Pool Study
- The analysis assesses whether any significant safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry casks for the reference plant as modeled, and the potential costs associated with such expedited transfer

# Data Used in the Regulatory Analysis

- Spent Fuel Pool Initiator Release Frequency
- Duration of On-site Spent Fuel Storage Risk
- Cost/Benefit Inflators
- Dollar per Person-Rem Conversion Factor
- Onsite Property Decontamination, Repair, and Refurbishment Costs
- Replacement Energy Costs
- Occupational Worker Exposure (Accident)
- Long-Term Habitability Criteria
- Other Key Data

# Assumptions used in the Regulatory Analysis

- Fuel Assembly Decay Heat as a Function of Burnup and Cooling Time
- Dry Storage Upfront Costs
- Incremental Costs Associated with Earlier Dry Storage Cask Purchase and Loading
- Incremental Annual Independent Spent Fuel Storage Installation Operating Costs
- Dry Storage Occupational Exposure (Routine)
- Number of Projected Dry Storage Casks Required

# Sensitivity Analysis

- Present Value Calculations
- Dollar per Person-Rem Conversion Factor
- Replacement Energy Costs
- Consequences Extending Beyond 50 Miles
- Combined Effect of Consequences Extending Beyond 50 Miles and Dollar per Person-Rem Conversion Factor

# Reference Plant Regulatory Analysis Results

- Total Cost to the Reference Plant
  - $47 million (using a 7-percent discount rate)
  - $42 million (using a 3-percent discount rate)
  - Range from $16 to $47 million (sensitivity analyses)
- Value of Benefits to the Reference Plant
  - $500,000 (using a 7-percent discount rate)
  - $700,000 (using a 3-percent discount rate)
  - Range from $500,000 to $43 million (sensitivity analyses)
- Costs to NRC
  - Were ignored to calculate the maximum potential benefit

# Reference Plant Decision Rationale

- Regulatory Analysis
  - Alternative considered does not achieve a cost-beneficial increase in public health and safety for the reference plant
  - The three sensitivity studies also showed that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases
- Backfit Analysis
  - Comparison to Safety Goal Policy Quantitative Objectives
    - No early fatalities predicted within 1 mile from site boundary which meets the individual early fatality risk goal
    - SFP accident represents 0.13% fraction of $1.84 \times 10^{-6}$ per year societal risk goal
  - Cost-justified criteria are not met when evaluating the averted accident consequences
    - Not met when evaluating the averted accident consequences within 50 miles of the site consistent with the regulatory framework
    - Not met for any of the discounted sensitivity cases that extend the analyses beyond 50 miles

# Expanded Regulatory Analysis For All Spent Fuel Pools

- **Objective is to expand the Spent Fuel Pool Study Regulatory Analysis (Appendix D) to all Spent Fuel Pools**
  - SFPS Reference Plant is based on a BWR Mark I with elevated SFP
  - Staff developing methodology to apply SFPS results to other reactors, including PWRs and new reactors

# Grouping/Sensitivity Studies

- **Spent Fuel Pool Grouping by Configuration/ Design**
  1. BWR Mark I / II with non-shared spent fuel pool (SFP) located well above grade
  2. PWR & BWR Mark III with non-shared SFP located at grade with at least one exposed side
  3. Advanced reactor SFPs
  4. Shared SFPs
  5. SFPs located below grade
  6. SFPs at decommissioned plants (fuel in pool)
  7. Decommissioned plants with fuel in ISFSI or shipped offsite
- **Sensitivity Studies**
  1. Consequences beyond 50 miles
  2. Population density
  3. Discount factors (7%, 3%, 2%)

# Regulatory Analysis Inputs

| Parameter | Low Est. | Base Case | High Est. |
|---|---|---|---|
| Site seismicity<br>• Bin 3 (SFPS F4)<br>• Bin 4 | $1.7 \times 10^{-5}$ (PB3)<br>$4.9 \times 10^{-6}$ | $1.7 \times 10^{-5}$ (PB3)<br>$4.9 \times 10^{-6}$ | $3 \times 10^{-5}$ (Brunswick)<br>$4.9 \times 10^{-6}$ |
| Ac power fragility | 1.0 (bounding) | 1.0 (bounding) | 1.0 (bounding) |
| Refueling freq. | 24 months | 18 and 24 months | 18 months |
| Liner fragility<br>• Bin 3 & 4 | 1.0 (bounding) | 1.0 (bounding) | 1.0 (bounding) |
| Insufficient nat. circ | 8.2 – 100% | 8.2 – 100% | 100% (bounding) |
| Flex mitigation likelihood | Higher success than SFPS | Same as SFPS or higher | Same as SFPS |
| **Source Term** | | | |
| SFP loading configuration | 1x4 immediately (PB3) | 1x4 immediately (PB3) | Uniform for 25d then 1x4 (assumed) |
| Release fraction | Based on previous studies | Based on previous studies | Based on previous studies |

# Regulatory Analysis Inputs (cont'd)

| Parameter | Low Est. | Best Case | High Est. |
|---|---|---|---|
| **Dose Consequence Analysis** | | | |
| Population density & demographics | 93 & 169 people/sq mi | 317 people/sq mi | 688 people/sq mi |
| Weather conditions & modeling | Same as SFPS (PB3) | Same as SFPS (PB3) | Same as SFPS (PB3) |
| Exposure & health effects modeling | 500 mrem annual - LNT | 2 rem first year, 500 mrem thereafter - LNT | 2 rem annual - LNT |
| Evacuation assumptions & modeling | Same as SFPS (PB3) | Same as SFPS (PB3) | Same as SFPS (PB3) |
| **Offsite Property Analysis** | | | |
| Economic data | Site specific using SECPOP2000) (Palisades) | Site specific using SECPOP2000) (Surry) | Site specific using SECPOP2000) (PB3) |

# Regulatory Analysis Alternatives

- Regulatory Baseline (1x4 high density loading)
- Low-Density Storage (1x4 for most recent discharged fuel)
- High-Density Storage (1x8, or other beneficial arrangement)
  - May require temporary increase in rate of transfer to dry cask storage
- Enhanced Mitigation Consistent with Storage
  - Further reduce the likelihood of spent fuel pool fires

# Next Steps

- Finish Tier 3 Phase 1 Analysis with consideration of stakeholder feedback
- ACRS Full Committee Meeting October 3, 2013
- Tier 3 Phase 1 Commission Paper Mid-October

# EXHIBIT 4

***Concerns Over Measurement of Fukushima Fallout***
**By DAVID MCNEILL | THE CHRONICLE OF HIGHER EDUCATION** MARCH 16, 2014
Photo


A decontamination worker at the entrance of Futaba, an abandoned town near the Fukushima nuclear plant. Credit Toru Hanai/Reuters

TOKYO — In the chaotic, fearful weeks after the Fukushima nuclear crisis began, in March 2011, researchers struggled to measure the radioactive fallout unleashed on the public. Michio Aoyama's initial findings were more startling than most. As a senior scientist at the Japanese government's Meteorological Research Institute, he said levels of radioactive cesium 137 in the surface water of the Pacific Ocean could be 10,000 times as high as contamination after Chernobyl, the world's worst nuclear accident.

Two months later, as Mr. Aoyama prepared to publish his findings in a short, nonpeer-reviewed article for Nature, the director general of the institute called with an unusual demand — that Mr. Aoyama remove his own name from the paper.

"He said there were points he didn't understand, or want to understand," the researcher recalled. "I was later told that he did not want to say that Fukushima radioactivity was worse than Chernobyl." The head of the institute, who has since retired, declined to comment for this article. Mr. Aoyama asked for his name to be removed, he said, and the article was not published.

The pressure he felt is not unusual — only his decision to speak about it. Off the record, university researchers in Japan say that even now, three years after the triple meltdown at the Fukushima Daiichi plant, they feel under pressure to play down the impact of the disaster. Some say they cannot get funds or university support for their work. In several cases, the professors say, they have been obstructed or told to steer clear of data that might cause public "concern."

"Getting involved in this sort of research is dangerous politically," said Joji Otaki, a biologist at Japan's Ryukyu University who has written papers suggesting that radioactivity at Fukushima has triggered inherited deformities in a

species of butterfly. His research is paid for through private donations, including crowdfunding, a sign, he said, that the public supports his work. “It’s an exceptional situation,” he said.

The precise health impact of the Fukushima disaster is disputed. The government has defined mandatory evacuation zones around the Daiichi plant as areas where cumulative dose levels might reach 20 millisieverts per year, the typical worldwide limit for nuclear-power-plant workers. The limit recommended by the International Commission on Radiological Protection is one millisievert per year for the public, though some scientists argue that below 100 millisieverts the threat of increased cancers is negligible.

In an effort to lower radiation and persuade about 155,000 people to return home, the government is trying to decontaminate a large area by scraping away millions of tons of radioactive dirt and storing it in temporary dumps. Experts at Japan’s National Institute of Advanced Industrial Science and Technology put the cost of this project at $50 billion — widely considered an underestimate.

The chance to study in this real-life laboratory has drawn a small number of researchers from around the world. Timothy A. Mousseau, a professor of biological sciences at the University of South Carolina who has written widely on Chernobyl, studies the impact of radiation on bird and insect life. He has published papers suggesting abnormalities and defects in some Fukushima species. But he said his three research excursions to Japan had been difficult.

In one case, a Japanese professor and two postdoctoral students dropped out of a joint research paper, telling him they could not risk association with his findings. “They felt it was too provocative and controversial,” he said, “and the postdocs were worried it could hamper their future job prospects.”

Mr. Mousseau is careful to avoid comparisons with the Soviet Union, which arrested and even imprisoned scientists who studied Chernobyl. Nevertheless, he finds the lukewarm support for studies in Japan troubling: “It’s pretty clear that there is self-censorship or professors have been warned by their superiors that they must be very, very careful,” he said.

The “more insidious censorship” is the lack of funding at a national level for these kinds of studies, he added. “They’re putting trillions of yen into moving dirt around and almost nothing into environmental assessment.”

Long before an earthquake and tsunami triggered the Fukushima meltdown, critics questioned the influence of Japan’s powerful nuclear lobby over the country’s top universities. Some professors say their careers have been hobbled because they expressed doubts about the nation’s nuclear policy and the coalition of bureaucrats, industrialists, politicians and elite academics who created it.

Mr. Aoyama, who now works at Fukushima University, sees no evidence of an organized conspiracy in the lack of openness about radiation levels — just official timidity. Despite the problems with his Nature article, he has written or co-written eight published papers since 2011 on coastal water pollution and other radiation-linked themes.

But stories of problems with Fukushima-related research are common, he said, including accounts of several professors’ being told not to measure radiation in the surrounding prefectures. “There are so many issues in our community,” he said. “The key phrase is ‘don’t cause panic.”’

He is also critical of the flood of false rumors circulating about the reach of Fukushima’s radioactive payload.

Ken Buesseler, a senior scientist at the Woods Hole Oceanographic Institution’s department of marine chemistry and geochemistry, in Massachusetts, who has worked with Mr. Aoyama, said he has spent much of his professional energy fighting the rumor mill. The cause is not helped, he added, by institutional attempts to gag Japanese professors.

“Researchers are told not to talk to the press, or they don’t feel comfortable about talking to the press without permission,” Mr. Buesseler said. A veteran of three post-earthquake research trips to Japan, he wants the authorities to put more money into investigating the impact on the food chain of Fukushima’s release of cesium and strontium. “Why isn’t the Japanese government paying for this, since they have most to gain?”

One reason, critics say, is that after a period of national soul searching, when it looked as if Japan might scrap its commercial reactors, the government is again supporting nuclear power. Since the conservative Liberal Democrats returned to power, in late 2012, Prime Minister Shinzo Abe has begun trying to sell Japan’s nuclear technology abroad.

Much of the government funding for academic research in Japan is funneled through either the Japan Society for the Promotion of Science or the Ministry of Education, Culture, Sports, Science and Technology. Proposals are screened by government officials and reviewed by an academic committee.

Yusuke Shoji, a spokesman for the ministry, cannot say how many proposals for studying the impact of radiation had been greenlighted, but he insists that the application system is fair. ‘‘The screening is conducted by peer review, so we don’t direct or don’t favor one particular research field,’’ he said. ‘‘We assess applications purely from the scientific point of view.’’ The Japan Society also says its applications process is not politicized.

Professors, meanwhile, say that rather than simply defend what is a piecemeal approach to studying the disaster, the government should take the lead in creating a large, publicly financed project.

"If we've ever going to make any headway into the environmental impact of these disasters, statistical power, scientific power, is what counts," said Mr. Mousseau of the University of South Carolina. "We get at it with massive replication, by going to hundreds of locations. That costs money."

***Correction: March 17, 2014***

*An earlier version of the headline with this article misstated the actions of the Japanese government. There are deep differences over how to determine the health impact of the Fukushima disaster. The authorities are not ''squelching" efforts to measure the effects of the accident.*

A version of this article appears in print on March 17, 2014, in The International New York Times.

# EXHIBIT 5

**Fukushima operator restarts water decontamination system**
24 MAR 2014

afp.com / Japanese government via Jiji Press /

Japanese inspectors look at the damaged building housing reactor number three at the Fukushima nuclear facility at Okuma, on June 17, 2011

The operator of Japan's crippled Fukushima nuclear plant said Monday it has switched on a key decontamination system that cleans radiation-tainted water used to cool the site's damaged reactors.

Last week, Tokyo Electric Power (TEPCO) said it had discovered a defect in its Advanced Liquid Processing System (ALPS) and switched it off for repairs.

The embattled firm said two of three lines that clean the toxic water were running again as of Monday afternoon.

A third line remained offline while workers tried to fix a filter defect which had prevented proper decontamination.

afp.com / TEPCO /

Leaked contaminated water is shown around a tank at TEPCO's Fukushima Dai-ichi nuclear power plant at Okuma in Fukushima prefecture, February 20, 2014

The problems meant supposedly purified water still had a large amount of strontium -- produced during nuclear reactions -- which accumulates in the bones and can cause several types of cancer in humans.

Company officials said there was no immediate safety risk, but added that the poorly-filtered water would have to be cleaned again.

It was not the first time the utility has switched off the system, which has been hit by a series of glitches since trial operations began a year ago.

TEPCO is struggling to handle a huge -- and growing -- volume of contaminated water at the tsunami-damaged plant. There are about 436,000 cubic metres of contaminated water stored at the site in about 1,200 purpose-built tanks.

Many experts say that at some point the water will have to be released into the sea after being scoured of the most harmful contaminants.

They say it will pose a negligible risk to marine life or people, but local fishermen and neighbouring countries are fiercely opposed.

# EXHIBIT 6

MONDAY, MAR 10, 2014 08:00 AM EDT

**Contaminated water still troubles Japan nuke plant**

MARI YAMAGUCHI, ASSOCIATED PRESS

Staff of Tokyo Electric Power Co. (TEPCO) and media walk in front of No. 1 reactors of Tokyo Electric Power Co. (TEPCO) at the crippled Fukushima No. 1 power plant in Okuma, Fukushima Prefecture on Mar. 10, 2014, nearly three years after the plant was paralyzed by the March 11 earthquake and tsunami in 2011. (AP Photo/Koji Sasahara,pool)(Credit: Koji Sasahara)

OKUMA, Japan (AP) — The radioactive water that has accumulated at Japan's crippled nuclear power plant remains the biggest problem hampering the cleanup process three years after the disaster.

The Fukushima Dai-ichi plant has stabilized substantially since the March 11, 2011, earthquake and tsunami destroyed its power and cooling system, triggering meltdowns. Massive amounts of water are being used to cool the melted cores at three reactors, but some of the contaminated water has seeped through the ground into the Pacific and leaked repeatedly from storage tanks.

Plant chief Akira Ono said Monday that improving water management is crucial not only to the plant cleanup but also decontamination of the area so evacuees can return to their homes.

"The most pressing issue for us is the contaminated water, rather than decommissioning," Ono said during a plant tour for foreign media, including The Associated Press. "Unless we resolve the problem, fear of the society continues and the evacuees cannot return home."

Experts say the water leaks are spreading radiation across the plant and into the sea, hampering the cleanup process. In order to mitigate the problem, TEPCO will build an underground ice wall around the four damaged reactor units to block contaminated water from leaking out while keeping underground water from flowing in — a multibillion-dollar government-funded project.

On Monday, workers were making final preparations to activate an experimental ice wall at a test site at the plant. The test is set to start within days, then the nearly 2 kilometer (1.2 mile) wall around the four units would be built for use sometime next year. A similar method has been used at a U.S. nuclear plant, but one with this magnitude is untested, and some experts say backup measures should be installed.

The plant has accumulated 436,000 tons of contaminated water stored in 1,200 industrial tanks that have taken over large parts of the plant.

Repeated water leaks, as well as preventive measures, monitoring and water have caused higher levels of exposures among workers.
The plant's operator, Tokyo Electric Power Co., acknowledged in July that contaminated underground water has been flowing into the ocean for some time, soon after the crisis began.
The disaster is the world's worst atomic accident since Chernobyl in 1986. More than 100,000 people have not returned home due to fear of radiation from the plant.

# EXHIBIT 7

Fukushima water decontamination suspended indefinitely
Published time: March 20, 2014 14:09
Edited time: March 21, 2014 08:59

Members of the media and Tokyo Electric Power Co. (TEPCO) employees, wearing protective suits and masks, walk toward the No. 1 reactor building at the tsunami-crippled TEPCO's Fukushima Daiichi nuclear power plant in Fukushima prefecture March 10, 2014.(Reuters / Koji Sasahara)

Treatment of radioactive water at Fukushima Daiichi nuclear power plant might be indefinitely suspended after malfunctions crippled the water purification process and recontaminated thousands of tons of partially purified water, Japanese media report.

The failure in the system, known as the Advanced Liquid Processing System (ALPS), is the latest setback in Tokyo Electric Power Co.'s (TEPCO) uphill battle to stockpile radioactive water, which is ballooning at a rate of 400 tons per day.

TEPCO said up to 900 tons of water, which had not been sufficiently cleaned in the ALPS equipment, flowed into a network of 21 tanks that were holding 15,000 tons of treated water. Not only have the 21 tanks been rendered unusable, but all 15,000 tons of previously cleaned water will now have to be retreated.

While efforts are underway to measure the full extent of the contamination, TEPCO officials said the problem was not noticed prior to March 18 because no abnormalities were detected in water sampled on March 14, Japan's Asashi Shimbun daily reports.

*"We never expected radioactive water to flow into the storage tanks,"* Masayuki Ono, acting general manager of TEPCO's Nuclear Power & Plant Siting Division, told the paper. *"We should have been better prepared. We have no idea how long it will take to clean them if we decided to do so."*

The ALPS system was developed to dramatically curb the radiation level of highly contaminated water that is accumulating at the plant. The APS consists of 14 steel cylinders through which the contaminated water is filtered. After the filtering, waste materials like the absorbent and remaining sludge are transferred to high-integrity containers (HICs) that are transported to a temporary storage facility.

The ALPS can remove 62 different types of radionuclides, including strontium and cobalt from contaminated water. While the system cannot remove tritium – a radioactive isotope of hydrogen – the purification of water through the system is expected to reduce damage levels if water leaks from storage tanks.

The equipment, which is supposed to be able to treat up to 750 tons of contaminated water a day, has been undergoing trial runs since March 2013. The system, however, has been plagued with problems from the outset. The

latest glitch and the subsequent recontamination was caused when one of the three ALPS lines failed to remove radioactive substances to a sufficient level.

Water from the March-17 sample water that was supposed to have been treated along one of the three channels of the ALPS system was discovered to still contain one-10th of the original concentration of radioactive substances. The system, however, is supposed to reduce that level to one-100,000th of the initial readings.

The finding prompted TEPCO to shut down ALPS operations along all three channels on March 18. In another incident, an ALPS pump stopped working in February, leaving only one of the two lines being tested at the time operational. With only one line working, the daily clean-up capacity dropped to one-third its capacity: 250 tons.

Approximately one week prior, around 100 tons of highly radioactive water leaked from one of the plant's tanks.

In mid-January, TEPCO warned that nuclear radiation at the boundaries of the damaged facility had jumped to eight times the government safety guidelines, while, only a week into the New Year, plant operators once again had to stop using its systems to decontaminate radioactive water. Compounding their problems at the time, a crane used to get rid of the container from the ALPS ceased functioning.

Meanwhile, a United Nations rights investigator said Japan should expand its cancer tests beyond the thyroid screenings being employed by local authorities, Bloomberg reports.

"Why don't we have a urine analysis, why don't we have a blood analysis?" said Grover, who also recommended that the tests be expanded to a broader geographical area. *"Let's err on the side of caution,"* Grover, a UN special rapporteur who surveyed the events surrounding the March 11, 2011 disaster, said in Tokyo on Thursday.

To date, 75 cases of thyroid cancer have been found among the 254,000 residents who had been tested as of February 7, the Asahi Shimbun reported at the time.

Only residents of Fukushima Prefecture who were18 or younger under at the time of the 2011 Fukushima nuclear disaster are eligible to receive the thyroid gland tests administered by the prefectural government.

Medical and government officials in Fukushima said they did not believe those 75 instances of cancer are linked to the 2011 disaster.

On March 11, 2011, a 9.0 megathrurst earthquake struck off the coast of Japan. The quake triggered a massive tsunami, which inundated the nuclear power plant causing three reactors to melt down. More than 18,000 people were killed across Japan, with entire communities destroyed or deemed uninhabitable.

# EXHIBIT 8

NUREG/CR-6451
BNL-NUREG-52498

# A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants

**Manuscript Completed: April 1997**
**Date Published: August 1997**

**Prepared by**
**R. J. Travis, R. E. Davis, E. J. Grove, M. A. Azarm**

**Brookhaven National Laboratory**
**Upton, NY 11973**

**G. J. Mencinsky, NRC Program Manager**

**Prepared for**
**Division of Regulatory Applications**
**Office of Nuclear Regulatory Research**
**U.S. Nuclear Regulatory Commission**
**Washington, DC 20555-0001**
**NRC Job Code L2590**

# ABSTRACT

An evaluation of the nuclear power plant regulatory basis is performed, as it pertains to those plants that are permanently shutdown (PSD) and awaiting or undergoing decommissioning. Four spent fuel storage configurations are examined. Recommendations are provided for those operationally based regulations that could be partially or totally removed for PSD plants without impacting public health and safety.

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# FIGURES

# TABLES

# EXECUTIVE SUMMARY

The long-term availability of less expensive power and the increasing plant modification and maintenance costs have caused some utilities to re-examine the economics of nuclear power. As a result, several utilities have opted to permanently shutdown their plants. Each licensee of these permanently shutdown (PSD) plants has submitted plant-specific exemption requests for those regulations that they believe are no longer applicable to their facility. The preparation and subsequent review of these exemption requests represents a large level of effort for both the licensees and the NRC staff. This experience has indicated the need for an explicit regulatory treatment of PSD nuclear power plants.

This report presents a regulatory assessment for generic BWR and PWR plants that have permanently ceased operation in support of NRC rulemaking activities in this area.

After the reactor vessel is defueled, the traditional accident sequences that dominate the operating plant risk are no longer applicable. The remaining source of public risk is associated with the accidents that involve the spent fuel. Previous studies have indicated that complete spent fuel pool drainage is an accident of potential concern. Certain combinations of spent fuel storage configurations and decay times, could cause freshly discharged fuel assemblies to self heat to a temperature where the self sustained oxidation of the zircaloy fuel cladding may cause cladding failure.

**Spent Fuel Configurations**

This study has defined four spent fuel configurations which encompass all of the anticipated spent fuel characteristics and storage modes following permanent shutdown. Spent fuel which (due to a combination of storage geometry, decay time, and reactor type) can support rapid zircaloy oxidation is designated as Spent Fuel Storage Configuration 1 - "Hot Fuel in the Spent Fuel Pool." Configuration 1 encompasses the period commencing immediately after the offload of the core to a point in time when the decay heat of the hottest assemblies is low enough such that no substantial zircaloy oxidation takes place (given the pool is drained), and the fuel cladding will remain intact (i.e., no gap releases).

After this point, the fuel is considered to be in Configuration 2 - "Cold Fuel in the Spent Fuel Pool." The fuel can be stored on a long-term basis in the spent fuel pool, while the rest of the plant is in safe storage or decontaminated (partial decommissioning). Alternatively, after decay heat loads have declined further, the fuel can be moved to an ISFSI (designated as spent fuel storage Configuration 3). This would allow complete decommissioning of the plant and closure of the Part 50 license. Spent fuel storage Configuration 4 assumes all spent fuel has been shipped offsite. This configuration assumes the plant Part 50 license remains in effect only because the plant has not been fully decontaminated and cannot be released for unrestricted public access.

A representative accident sequence was chosen for each configuration. Consequence analyses were performed using these sequences to estimate onsite and boundary doses, population doses and economic costs.

**Regulatory Assessment**

After a plant is permanently shutdown, awaiting or in the decommissioning process, certain operating based regulations may no longer be applicable. A list of candidate regulations was identified from a screening of 10 CFR Parts 0 to 199. The continued applicability of each regulation was assessed within the context of each spent fuel storage configuration and the results of the consequence analyses. The regulations that are no longer fully applicable to the permanently shutdown plant are summarized below:

The set of regulations that are designed to protect the public against full power and/or design basis accidents are no longer applicable and can be deleted for all spent fuel storage configurations of the permanently shutdown plant. These regulations include combustible gas control (50.44), fracture prevention measures (50.60, 50.61), and ATWS requirements (50.62).

Other regulations, although based on the operating plant, may continue to be partially applicable to the permanently defueled facility. This group of requirements includes the Technical Specifications (50.36, 36b), the fire protection program (50.48) and Quality Assurance (50.54(a) and Part 50 Appendix B).

The requirements for emergency preparedness (50.47, 50.54(q) and (t), and Part 50 Appendix E), onsite property damage insurance (50.54(w)) and offsite liability insurance (Part 140), were evaluated using the accident consequence analysis. Since the estimated consequences of the Configuration 1 representative accident sequence approximate those of a core damage accident, it is recommended that all offsite and onsite emergency planning requirements remain in place during this period, with the exception of the Emergency Response Data System requirements of Part 50, Appendix E. Subject to plant specific confirmation, the offsite emergency preparedness (EP) requirements are expected to be eliminated for Configuration 2, on the basis of a generic boundary dose calculation. Part 50 offsite EP requirements can also be eliminated for Configurations 3 and 4 because the spent fuel has been transferred to an ISFSI (subject to Part 72 requirements) or transported offsite. Without spent fuel, the plant is not a significant health risk. It is recommended that the onsite property damage and the offsite liability insurance levels remain at operating reactor levels for the duration of Configuration 1. The consequence analyses support reduced insurance requirements for the remaining configurations (2,3, and 4).

# FOREWORD

The information in this report is being considered by the U.S. Nuclear Regulatory Commission (NRC) staff in the development of amendments to its regulations for permanently shutdown nuclear power reactors in the process of decommissioning. The NRC has undertaken a number of initiative to reduce the regulatory burden for licensees that are in the process of permanently removing nuclear facilities from service. This report provides baseline data to the NRC for evaluating which regulations may be considered for amending to enhance the regulatory effectiveness during decommissioning.

Dr. Sher Bahadur, Chief
Regulation Development Branch
Division of Regulatory Applications
Office of Nuclear Regulatory Research

# ACKNOWLEDGEMENTS

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# ACRONYMS

| | |
|---|---|
| AC | alternating current |
| AEC | Atomic Energy Commission |
| ANI | American Nuclear Insurers |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| ASME | American Society of Mechanical Engineers |
| ATWS | anticipated transient without scram |
| BNL | Brookhaven National Laboratory |
| BWR | boiling water reactor |
| BWST | borated water storage tank |
| C | Celsius |
| DECON | decontamination |
| DOE | Department of Energy |
| ECCS | emergency core cooling systems |
| EDE | effective dose equivalent |
| EDG | emergency diesel generator |
| EOF | emergency operations facility |
| EP | emergency planning and preparedness |
| EPA | Environmental Protection Agency |
| EPRI | Electric Power Research Institute |
| EPZ | emergency planning zone |
| EQ | environmental qualification |
| FSAR | final safety analysis report |
| GWD | gigawatt day |
| H | high release factors |
| HLW | High Level Waste |
| HVAC | heating, ventilation and air conditioning |
| IAEA | International Atomic Energy Agency |
| ISFSI | independent spent fuel storage installation |
| ISI | inservice inspection |
| IST | inservice testing |
| KW | kilowatt |
| L | low release factors |
| LOCA | loss of coolant accident |
| LWR | light water reactor |
| MACCS | Melcor Accident Consequence Code System |
| MAERP | Mutual Atomic Energy Reinsurance Pool |
| MRS | monitored retrievable storage |
| MTU | metric ton of uranium |
| MW, MWe | electrical megawatt |
| NEIL | Nuclear Electric Insurance Limited |

# ACRONYMS (Cont'd)

| | |
|---|---|
| PAGs | protective action guides |
| PNL | Pacific Northwest Laboratory |
| POL | possession only license |
| PRA | probabilistic risk assessment |
| PSD | permanent shutdown |
| PWR | pressurized water reactor |
| QA | quality assurance |
| RG | regulatory guide |
| RPS | reactor protection system |
| RSS | Reactor Safety Study |
| SAFSTOR | safe storage with deferred decontamination |
| SBO | station blackout |
| SSCs | structures, systems and components |
| SFP | spent fuel pool |
| SFUEL1W | a spent fuel heatup code |
| SHARP | spent-fuel heatup code |
| SMN | special nuclear material |
| SMUD | Sacramento Municipal Utility District |
| SNL | Sandia National Laboratory |
| SRO | senior reactor operator |
| TMI | Three Mile Island |
| TSC | Technical Support Center |
| US | United States |

# 1 INTRODUCTION

The long-term availability of less expensive power, compounded by the increasing plant modification and maintenance costs, have caused some utilities to re-examine the economics of nuclear power. As a result, several plants with years, some with decades, left on their operating licenses have opted to permanently shutdown their facilities.

At present, six (6) nuclear power plants* are permanently shutdown in various stages of the decommissioning process. The absence of a clearly defined regulatory path for these licensees has become apparent. Each of the permanently shutdown (PSD) licensees has submitted plant-specific exemption requests for those regulations that they believe are no longer applicable to their facility. The lack of a regulatory roadmap for the permanently defueled plant has resulted in a large effort for both the licensees and the NRC staff attributable to the development and review of plant-specific exemption requests. This experience has established the need for an explicit regulatory treatment of PSD nuclear power plants, including:

- the clarification of the regulations for decommissioning nuclear power plants,
- the activities that are permissible for major phases of the decommissioning process,
- the specification of those Part 50 regulations that are applicable only to plants authorized to operate.[1]

Brookhaven National Laboratory (BNL) has undertaken a program (FIN L-2590) "Safety and Regulatory Issues Related to the Permanent Shutdown of Nuclear Power Plants Awaiting Decommissioning," to support the last NRC goal stated above, i.e., "to determine the extent and types of safety criteria that should remain as part of the decommissioning regulations to assure that the health and safety of public is protected when a licensee enters the permanent shutdown condition in preparation for plant decommissioning."

This NUREG/CR documents the results of this program.

The remainder of this report is structured as follows:

Section 2, "Background" presents a brief discussion of the changes that are likely to take place when a licensee permanently ceases operation of a nuclear power plant. As the primary source of public risk, the focus of this discussion is the storage alternatives for the spent fuel. This section, in conjunction with Appendix A, "Previous Examinations of Spent Fuel Pool Accidents," also summarizes the assumptions and conclusions of earlier studies in this area. This information can be helpful as it provides the necessary context for the assessment of the present study's assumptions and conclusions.

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*Fort St. Vrain, Rancho Seco, San Onofre Unit 1, Three Mile Island 2, Trojan, and Yankee Rowe, are undergoing decommissioning. Shoreham has completed the process and the license has been terminated.

Section 3, "Input Assumptions," provides detailed information and assumptions (such as accident initiator, timing, source terms, meteorology, population, etc.) that are necessary to support the accident consequence analyses. In support of potential rulemaking, the calculation assumptions of Section 3 have been developed to envelope the end of life plant shutdowns that are anticipated in the future. Thus, this study considers spent fuel pools that are full to capacity with high burnup fuel, and an offsite population density that is consistent with end of plant life. As such, these assumptions tend to be conservative with regard to those plants that are currently shutdown awaiting, or in various stages of, decommissioning.

Section 4, "Results of the Consequence Analyses," presents the estimated accident consequences for each spent fuel storage configuration, including societal dose, condemned land area, and accident cost. Multiple cases were evaluated using different inventory and source-term assumptions. BNL has chosen a "best estimate" case for each configuration.

Section 5, "Regulatory Assessment Summary," and Appendix B present the evaluation of the current operating plant based NRC regulations as applied to the permanently shutdown nuclear power plant. The applicability of each candidate regulation is assessed for each spent fuel storage configuration, based on the likely status of the physical plant and the consequence analysis of the preceding section.

Sections 6 and 7, respectively, summarize the report and provide the necessary references.

# 2 BACKGROUND

Once a decision is made to permanently cease operation of its nuclear power plant, the licensee will defuel the reactor vessel. In parallel (or perhaps in anticipation of permanent shutdown) the licensee will apply for an NRC license amendment to withdraw the authority to operate the plant. It also provides a basis to remove the regulatory requirements that are no longer necessary to protect the health and safety of the public. Thus, the amendment to remove the authority to operate provides a basis for a licensee to begin eliminating personnel, equipment, and activities pursuant to 10CFR 50.59 analyses, license amendments and exemption requests. The regulatory ambiguity regarding the permanently shutdown nuclear power plant has prompted the NRC to develop further guidance in this area*. However, the basis for any regulatory relief must ultimately address the potential impact on public health and safety. Previous decommissioning studies[3-4] have shown that the offsite doses associated with decommissioning accidents that do not involve spent fuel are negligible. Therefore, this study has focused on the spent fuel storage alternatives after a plant has been permanently shutdown and the potential public risk associated with each alternative.

After the reactor vessel is defueled the traditional accident sequences that dominated the operating plant risk are no longer applicable. The remaining source of public risk is associated with the accidents that involve the spent fuel stored in the spent fuel pool (SFP). As discussed in Appendix A, accidents involving spent fuel, although limited to the 1/3 core offloads associated with refueling were considered as part of the spectrum of nuclear power plant risk as early as the Reactor Safety Study (WASH 1400). More recently, Sandia National Laboratories (SNL) studies[5-6] have indicated that complete spent fuel pool drainage, with certain combinations of spent fuel storage configurations and decay times, could cause freshly discharged fuel assemblies to self heat to a temperature where the oxidation of the zircaloy fuel cladding may become self sustaining. Follow-up efforts by BNL[7] applied simplified PRA analyses to quantify the frequency of initiating events that could compromise the SFP integrity; the conditional probability of subsequent system failure, fuel failure probability; the magnitude of radionuclide releases to the environment and the consequences of those releases.

A 1989 BNL report,[8] describes a value/impact assessment of various proposed options intended to reduce the risk posed by potential accidents occurring in commercial nuclear power plant spent fuel pools. As was the case with previous efforts, attention was limited to an operating plant. The risk dominant accidents, source terms and inventory considered in this later effort were identical to those investigated by Sailor, et al. in Reference 7. Major differences in the estimation of the off-site consequences exist between these two studies which are primarily attributable to the higher population density assumptions of the later report.

This study has defined four (4) spent fuel configurations which encompass all anticipated spent fuel characteristics and storage modes following permanent shutdown. Spent fuel which, due to a combination of storage geometry, decay time, and reactor type, can support rapid zircaloy oxidation is designated as Spent Fuel Storage Configuration 1 - "Hot Fuel in the Spent Fuel Pool." Configuration 1 encompasses the period commencing immediately after the offload of the core to a point in time when the decay heat

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*Although a licensee is prohibited from making changes that materially affect costs, methods, or options for decommissioning the facility, the extent of permissible decommissioning activities has been clarified by issuance of final rule (61 FR 39278) amending regulations on decommissioning procedures.

of the hottest assemblies is low enough such that no zircaloy oxidation takes place, and the fuel cladding will remain intact (i.e., no gap releases).

At this point the fuel is considered to be in Configuration 2 - "Cold Fuel in the Spent Fuel Pool." The fuel can be stored on a long-term basis in the spent fuel pool, while the rest of the plant is in SAFSTOR* or decontaminated (partial decommissioning). Alternatively, after decay heat loads have declined further,** the fuel can be moved to an ISFSI (designated as spent fuel torage Configuration 3). This would allow complete decommissioning of the plant and closure of the Part 50 license.

Given the present unavailability of a permanent geological high level waste repository, or an interim Monitored Retrievable Storage (MRS) facility the fuel is expected to remain onsite for an indefinite time period.

At some point in the future, a MRS facility or a high level waste repository will become available. Spent fuel storage Configuration 4 assumes all spent fuel has been shipped offsite. This configuration assumes the plant Part 50 license remains in effect only because the plant has not been fully decontaminated and cannot be released for unrestricted public access.

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*Safe storage followed by deferred decontamination.

**Limits are placed on the burnup, decay time, enrichment and decay heat of the spent fuel assemblies to ensure the ISFSI design heat load is not exceeded. Although 10CFR Part 72 specifies a minimum of one year pool decay time, plant ISFSI technical specifications specify minimum decay times up to 10 years.

# 3 SPENT FUEL STORAGE CONFIGURATION INPUT ASSUMPTIONS

The purpose of this section is to define the input assumptions for each spent fuel storage configuration to support the consequence analyses of the next section. A set of assumptions was developed that is used in Section 4 to provide an estimate of the accident consequences that envelope future end of life nuclear power plant shutdowns, as well as plants that have prematurely ceased operation. However, an effort has been made to avoid unduly pessimistic assumptions or combinations of assumptions. The accident consequences thus obtained, are believed to be reasonably bounding for present and future closures and are not so overly conservative as to clearly represent some high (but unspecified) percentile result.

The input assumptions for each configuration will be discussed for PWRs and BWRs, respectively. Table 3.1 presents a summary of this section.

## 3.1 Configuration 1 - Hot Fuel in the Spent Fuel Pool

Spent fuel storage Configuration 1 commences immediately after the permanently shutdown facility has completed the reactor vessel defueling. This configuration models the potential consequences of rapid zircaloy oxidation resulting from an event which has caused the draining of the spent fuel pool. After a suitable time period, dependent on assembly burnup and racking geometry, the decay heat is low enough to preclude the rapid oxidation phenomenon. The end of this configuration is defined as that point in time when the fuel decay heat is low such that the cladding remains intact upon extended exposure to the air.

The consequence analysis input assumptions for Configuration 1 are provided below in the form of generic PWR and BWR plant configurations.

### 3.1.1 Representative Plant and Fuel Pool Data

The representative PWR* chosen for this study is a single 1130 MWe unit with 193 assemblies in the core. The corresponding 1155 MWe BWR has 764 assemblies. In accord with the industry trend to maximize storage capacity, both plants have high density fuel racking geometries.** The PWR spent fuel racks have a 10.40 inch cell to cell pitch and a five inch orifice at the bottom of each cell.[9] The BWR spent fuel racks a 6.255 inch pitch. Each BWR cell has a 4-inch orifice.[10] Variation in these parameters exist among various rack designs and manufacturers. These values were chosen to represent typical attributes.

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*The representative PWR and BWR geometries and spent fuel data were developed from a review of a limited set of plant information. They are generally the most conservative values from that set of information and are viewed as reasonably conservative, but not necessarily the most limiting configurations.

**Previous studies of the spent fuel rapid oxidation phenomenon have assumed a low density racking configuration for BWRs. (See Appendix A).

The spent fuel pool storage capacities were 1460 intact assemblies for the generic PWR and 3300 assemblies for the generic BWR. These are the average pool capacities of the current 193 assembly PWRs and 764 assembly BWRs. In order to envelope end of life shutdowns, this analysis assumed that the pools are full. The last full core offload was assumed to contain high burnup fuel (60,000 and 40,000 megawatt days per metric tons of heavy metal (MWD/MTU), PWR and BWR, respectively), to reflect the current trend to increase burnup. The earlier refueling discharges began at 20,000 MWD/MTU and increased linearly with each subsequent discharge to the ultimate assumed burnup. Consistent with Regulatory Guide 4.7, an exclusion boundary of 0.4 miles was assumed for each plant.

### 3.1.2 Accident Initiator and Timing

The accident initiator was a composite of events that can cause draining or boiloff of the spent fuel pool and expose the relatively hot spent fuel assemblies to an air environment. The initiator includes beyond design basis seismic events, spent fuel cask drop events, and other less dominant events such as spent fuel pool loss of cooling/makeup.

The composite initiator frequency of 2E-6 (PWR) and 7E-6 (BWR) events per year is adapted from the NUREG-1353 "best estimate" with modifications to reflect a higher spent fuel cask drop contributor associated with a higher assumed spent fuel transfer rate for the permanently shutdown plant. For the purposes of the offsite liability insurance discussion in Appendix B, the initiator frequency is equivalent to the release frequency.

The accident timing considered the minimum in-core decay requirements of the Standard Technical Specifications (about 4 days) and industry experience of several weeks to fully offload a core during refueling outages. For this study, the Configuration 1 accident initiator was assumed to occur 12 days following final shutdown.

### 3.1.3 Critical Decay Time

Previous studies[5-7] have defined the critical decay time as the duration, measured with respect to reactor shutdown, when the most recently discharged set of fuel assemblies have sufficient decay heat, that if the fuel pool were to completely drain, would heat to the point that clad oxidation would become self sustaining and eventually result in extensive clad failure with fission product release. This time is a function of the reactor type, spent fuel storage rack geometry and fuel burnup.

To be conservative, this effort chose to examine high density rack geometries for both PWR and BWR plants. In the time frame of the previous studies, high density racking was not widely used by in BWR plants. The previous efforts, therefore, do not provide results for this case.

The PWR high density racking geometry with a 5-inch orifice (albeit with low burnup fuel) was examined in NUREG/CR-4982. A 700 day critical decay time was estimated, using the SFUEL1W[5,6] code. based on a minimum decay power of 6 KW/MTU.

## Table 3.1 Spent Fuel Storage Configuration Matrix

| Parameters | Configuration 1 Hot Fuel in Spent Fuel Pool | | Configuration 2 Cold fuel in spent fuel pool | | Configuration 3 Spent fuel stored in ISFSI | Configuration 4 All spent fuel removed from site |
|---|---|---|---|---|---|---|
| | PWR | BWR | PWR | BWR | | All Fuel Removed |
| Representative Plant Data | Single Unit<br>1130 MWe<br>193 Fuel Assemblies | Single Unit<br>1155 MWe<br>764 Fuel Assemblies | *Same as Config. 1* | *Same as Config. 1* | N/A | N/A |
| Spent Fuel Pool Storage Cask Data | Capacity 1460 intact fuel assemblies<br>All slots filled<br>*Cask laydown area in pool* | Capacity 3300 intact fuel assemblies<br>*All slots filled*<br>Cask laydown area in pool | Same as Config. 1 | Same as Config. 1 | Storage Casks (metal & concrete)<br>28 PWR Fuel assemblies<br>56 BWR Fuel assemblies<br>NUHOMS<br>24 PWR Fuel assemblies<br>52 BWR Fuel assemblies | N/A |
| Spent Fuel Storage Rack Design Data | 2 Region design<br>Stainless steel max capacity<br>10.40 inch pitch<br>5 inch dia. bottom orifice hole (each cell) | Single region<br>Stainless steel max capacity<br>6.25 inch pitch<br>4 inch dia. bottom orifice hole (each cell) | Same as Config. 1 | Same as Config. 1 | N/A | N/A |
| Fuel Assembly Burnup | Burnup range 20-60 GWD/MTU | *Burnup range 20-40 GWD/MTU* | *Same as Config. 1* | *Same as Config. 1* | 5 year fuel decay time prior to storage in ISFSI<br><br>High burnup fuel placed in ISFSI (60 GWD/MTU for PWR, 40 GWD/MTU for BWR) | N/A |
| Accident Initiator | Spent fuel pool fire resulting from a spent fuel pool draindown | Same as PWR | Fuel handling accident resulting in gap release | | Tornado driven missile which results in breach of both primary & secondary seals, fuel cladding failure & fission product release | No credible decommissioning accidents can be postulated that have significant health effects |
| Accident Timing | 12 days after final reactor shutdown | | *3.5 years after final shutdown* | *2 years after final shutdown* | *5 years after final shutdown* | *5 years after final shutdown* |
| Exclusion Zone | 0.4 miles | | Same as Config. 1 | | 100 meters | N/A |
| Offsite Meteorological Data | Mean weather attributes from continental U.S. | | Same as Config. 1 | | A high wind speed is assumed to approximate a tornado's dispersion | N/A |
| Population Distribution | Population density of 1000 persons/sq. mi. out to 30 miles. | | Same as Config. 1 | | Same as Config. 1 | N/A |
| Accident Inventory and Source Term | Four accident inventories are examined ranging from fire involving full pool to gap release of last core offload. High (H) and low (L) release fractions based on NUREG/CR-4982 as modified by more recent studies of gap release and high burnup fuel are examined for each inventory. | | Accident inventory is one fuel assembly. High and low gap release fractions examined | | Two accident inventories examined: one assembly or entire cask. High and low gap releases for each | N/A |

It should be stressed that there are uncertainties associated with this SFUEL1W calculation. The authors of the present study fully agree with the code limitations presented in NUREG/CR-4982 report. The SFUEL1W code provides a stylized analysis of the progression of events following the complete loss of spent fuel pool coolant and as such, does not have the ability to realistically model actual spent fuel pool configurations.

In response to the need to accurately predict the likelihood of reaching critical clad temperatures with realistic spent fuel pool configurations, BNL has developed the SHARP code (Spent-fuel Heatup: Analytical Response Program.)[45]

This code has been used, in conjunction with the Configuration 1 spent fuel data from Table 3.1 to develop maximum clad temperature as a function of decay time, given a loss of all spent fuel pool water. These relationships are presented as Figures 3.1 and 3.2 for the PWR and BWR representative geometries.

The end of Configuration 1 has been defined as the decay time that is necessary to ensure that the fuel rod cladding remains intact given a loss of all spent fuel pool water. The previous study[7] defined 650°C as a maximum temperature for cladding integrity. The Workshop on Transportation Accident Scenarios[47] estimated incipient clad failure at 565°C with expected failure at 671°C, presumably based on expert opinion. Given that the large seismic event is the dominant contributor to the configuration 1 initiator, it is likely that it would take a prolonged period of time to retrieve the fuel, repair the spent fuel pool or establish an alternate means of long-term spent fuel storage. Therefore, we presume there will be a significant period of time that the fuel will be exposed to air. On this basis, BNL has chosen a temperature of 565°C as the critical cladding temperature. This results in critical decay times of about 17 months for the representative PWR and 7 months for the representative BWR.

### 3.1.4 Meteorological and Population Data

Weather and its variability play an important role in the estimation of consequences that may result from a release of radioactivity to the environment. The prevailing weather conditions at the time of release will influence: the extent of downwind transport and lateral dispersion; the atmospheric concentration; and the extent and severity of land contamination. The SNL Siting Study, NUREG/CR-2239[13] and a BNL reassessment[14] were utilized to develop a representative meteorology for the continental United States composed of: mean weather attributes (wind speed, stability, class occurrence total hours, and amount of rain for Omaha, NB); a generic mean wind rose; and an average mixing height.

This study has adopted a generic population distribution within a 500 mile radius of the site that will reasonably envelope the majority of the current reactor sites* and account for future population growth over the life of the plant.

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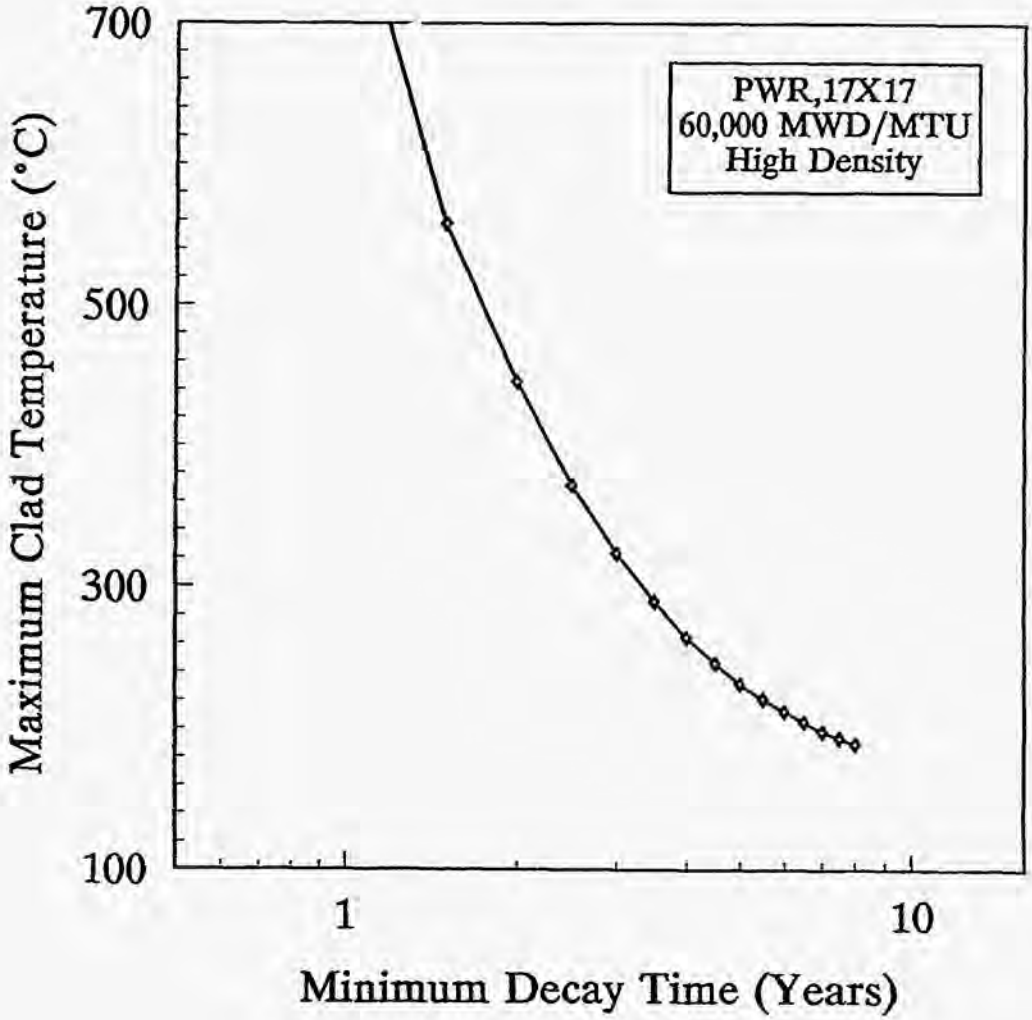
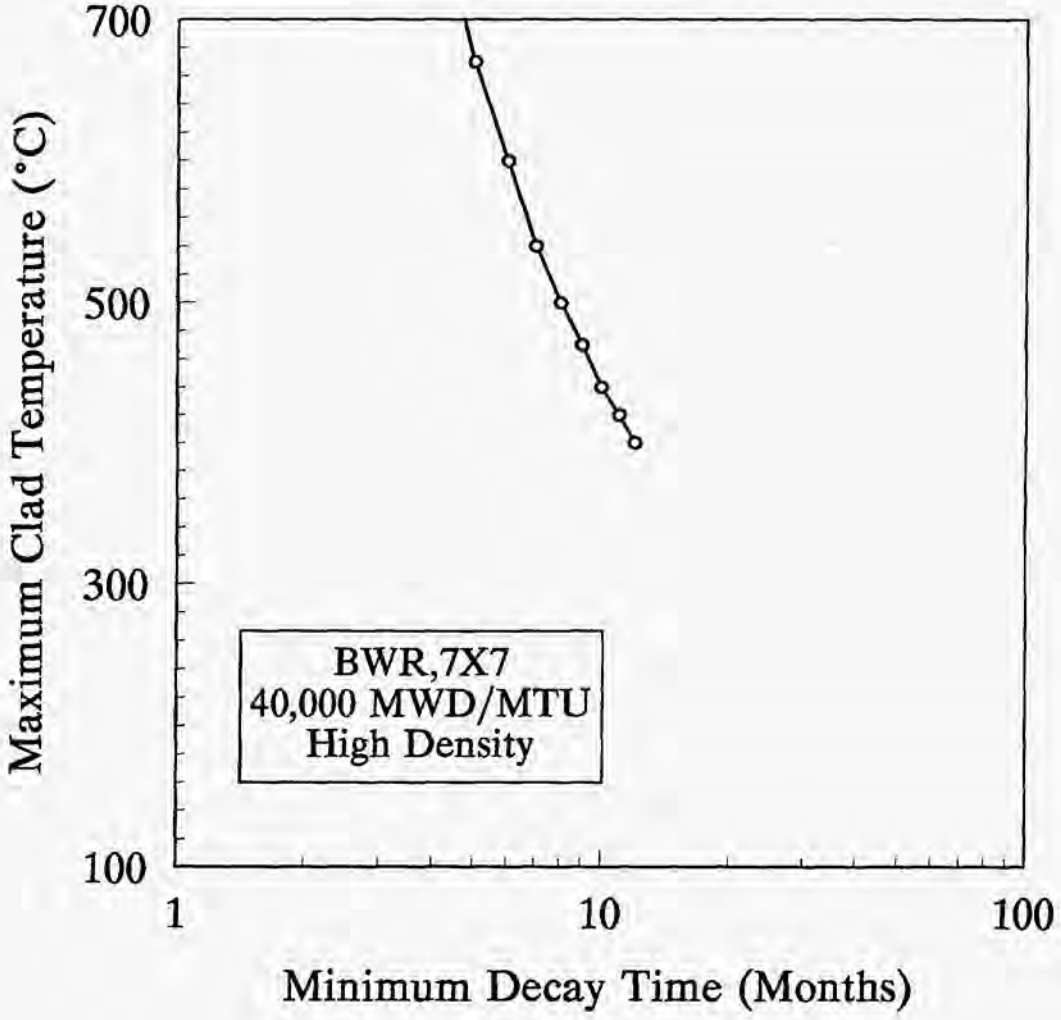
*There are several existing plant sites (i.e., Indian Point, Limerick, and Zion) that precede the issuance of R.G. 4.7 and exceed the site population distributions generally considered acceptable by current NRC policy.

A uniform population distribution (0-30 miles) of 1000 persons per square mile has been specified based on the end of life average population density from Regulatory Guide 4.7. Between 30 to 50 miles, we have assumed a large city of 10 million and a uniform population density of 280 persons/mile$^2$ for the remaining land in this region.[14] A uniform population density of 200 persons/mile$^2$ (twice the current average of the 48 contiguous states) was assumed for the area 50 to 500 miles from the plant.




*(Adapted from Reference 46)*

**Figure 3.1 Spent fuel temperature as a function of time for the representative PWR configuration**




*(Adapted from Reference 46)*

**Figure 3.2 Spent fuel temperature as a function of time for the representative BWR configuration**

### 3.1.5 Accident Inventory and Source Term

The spent fuel pool inventory at accident initiation is a function of the ages and burnups of the spent fuel discharges that occurred over the life of the plant. The DOE High Level Radioactive Waste Management Database[15] was used as the source of the generic spent fuel inventory data for discharges one year or older.

The inventory of material at risk 12 days after reactor shutdown (i.e., at the beginning of Configuration 1) was developed from both the DOE Spent Fuel Data Base and the default reactor core inventories provided in the MELCOR Accident Consequence Code System (MACCS).

MACCS Version 1.5.11.1[16-17] was used in the next section to model the postulated accident consequence. Like other consequence codes, MACCS models radionuclide releases that occur shortly after reactor shutdown. The code has a default set of risk dominant radionuclide species that is consistent with the premise of a release within days of shutdown. In contrast, the inventory of the spent fuel pool, including the last core offload, has had sufficient time for the short lived isotopes, which have important dose contributions, to decay away. The concern is that perhaps the MACCS default set of isotopes might not accurately model long lived isotopes that are relatively insignificant for short-term releases, but rise in prominence for spent fuel pool accidents. The code default isotopes set was spot checked with the DOE database[15] inventory for two offloads. It was determined that the MACCS code will capture greater than 90% of the activity in the spent fuel. Therefore, it was not necessary to revise the code's default isotope set to include any additional radionuclide species.

The atmospheric source term is a set of characteristics describing the radionuclide release to the environment. These characteristics include: the number of plume segments released, the associated timing duration and release height of each segment, the emergency response warning time and the radionuclide release fractions.

This study examined four cases for Configuration 1. The assumptions for each case are described below:

Case 1 Complete draining of the spent fuel pool occurs twelve days after shutdown. Rapid cladding oxidation starts in the last full core discharge and propagates throughout the pool.

Case 2 Complete pool drainage occurs, again at twelve days. The rapid zircaloy oxidation is limited to the last full core discharge (plus the last refueling offload for PWRs).

Case 3 Complete pool drainage occurs one year after shutdown. The lowered decay heat does not cause rapid oxidation, however the assemblies reach high temperatures and 50 percent of the fuel rods in the pool fail, resulting in a gap release.

Case 4 Partial pool drainage occurs at twelve days, exposing the upper portion of the fuel assemblies. This case assumes all fuel rods in the last full core discharge experience cladding failure, again resulting in a gap release.

This study used the release fractions of NUREG/CR-4982, as modified by studies associated with gap inventory and high burnup fuel.[18-22,33] Table 3.2 provides the source terms developed for the present study.

The majority of the high release fractions for Cases 1 and 2 were largely adopted from NUREG/CR-4982. However, the lanthanum (La) and cerium (Ce) groups have been adjusted slightly to reflect the observed release of fuel fines as part of the gap release in high burnup fuel. The low release fractions for Cases 1 and 2 assumed a decontamination factor (DF) of 10 for all fractions

**Table 3.2 Configuration 1 Release Fractions**

| Case | Release Characterization | NG | I | Cs | Te | Sr | Ru | La | Ce | Ba |
|---|---|---|---|---|---|---|---|---|---|---|
| 1H, 2H | Fire/High release | 1.0 | 1.0 | 1.0 | 2E-2 | 2E-3 | 2E-5 | 6E-6 | 6E-6 | 2E-3 |
| 1L, 2L | Fire/Low release | 1.0 | 0.5 | 0.1 | 2E-3 | 2E-4 | 2E-6 | 6E-7 | 6E-7 | 2E-4 |
| 3H, 4H | Gap/High release | 0.4 | 3E-2 | 3E-2 | 1E-3 | 6E-6 | 6E-6 | 6E-6 | 6E-6 | 6E-6 |
| 3L, 4L | Gap/Low release | 0.4 | 3E-3 | 3E-3 | 1E-4 | 6E-7 | 6E-7 | 6E-7 | 6E-7 | 6E-7 |

except noble gases and iodine. Cases 3 and 4 heat the fuel cladding to failure, but do not result in fire. The gap release fractions developed for this work differs markedly from the previous efforts. The noble gas fraction, 0.4, was based on high burnup/high linear power calculation and is therefore believed to be conservative. The fractions for the cesium (Cs), iodine (I), and tellurium (Te) groups were based on experimental observation. In the case of the high gap release, these were increased by a factor of ten to reflect evidence that these fractions may increase for high burnup fuels. For both the high and low gap releases, the Te fractions were corrected for the interaction observed to occur with the cladding, since unoxidized cladding will be present. The fractions for the remaining groups are established by the release of fuel fines.

For the set of low gap releases (Cases 3 and 4), all release fractions were reduced by an order of magnitude (DF=10) with the exception of noble gases (NG).

### 3.1.6 Emergency Response and Other Data Requirements

The MACCS code can model various emergency response actions such as evacuation, sheltering, and post accident relocation (including dose criteria). Consistent with NUREG/CR-5281,[8] this study assumed a short-term emergency response of no planned evacuation, followed by relocation at one day if projected doses are unacceptable. Long-term protective actions include permanent relocation, crop interdiction,and land decontamination or condemnation. The dose threshold for these actions are the MACCS default values which were also utilized in NUREG-1150.[23]

The code also considers land usage and economic data for the region surrounding the reactor site to estimate accident cases. The national average value of farmland of $2094/hectare and a mean value of $73,750/person for non-farm wealth was assumed.[23] The Omaha, Nebraska region, also used for the

mean meteorology, was used to model the code's agricultural data block, including the growing season and the fraction of land used for farming.

These estimated accident costs will be used to analyze the insurance issues for permanently shutdown nuclear power reactors.

## 3.2 Configuration 2 - Cold Fuel in the Spent Fuel Pool

Spent fuel storage Configuration 2 models the continued storage of the fuel in the spent fuel pool. Time has reduced the decay heat, and the rapid clad oxidation or clad rupture events of Configuration 1 are not likely. This section summarizes the input assumptions, such as accident initiator, and source terms that differ from those of the previous section. Other parameters (i.e., spent fuel pool data, rack design, and fuel burnup) remain consistent with the Configuration 1 baseline. A summary of each spent fuel configuration is provided in Table 3.1.

### 3.2.1 Accident Initiator and Timing

By definition, Configuration 2 eliminates the pool drainage accident scenarios of Configuration 1 from consideration. The prolonged exposure of the low-decay heat fuel in air is not expected to cause fuel rod clad failures. BNL has adopted the traditional fuel handling accident analysis of Regulatory Guide 1.25, with modifications. The present study assumed a single assembly is dropped in the spent fuel pool, resulting in damage to 100 percent of the rods in the affected assembly.

The estimated initiator frequency of 3E-4 events per year* was developed from industry refueling outage data reported in Reference 48, modified to reflect a higher assumed spent fuel transfer rate.

The accident was assumed to occur after the transition from Configuration 1, one to two years after final reactor shutdown.

### 3.2.2 Accident Inventory and Source Terms

The accident inventories for the Configuration 2 accident cases consist of a single two year old PWR fuel assembly or a single one year old BWR assembly. As before, the DOE spent fuel database**[15] was used to assemble the isotope quantities for the MACCS default set of nuclides.***

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*This is also the estimated release frequency.

**At 60,000 and 40,000 MWD/MTU burnup for the PWR and BWR cases, respectively.

***In both reactor types the MACCS default, risk dominant nuclides represent about 89 percent of the total activity in the fuel.

The source term is composed of the single assembly gap release. In addition to partial releases of the noble gases and iodine (if present), small releases of the remaining nuclide groups are expected on the basis of experimentally observed releases of fuel fines. The Configuration 2 high gap release fractions are the same as Case 3H of Table 3.2 in the previous section. The low gap source term assumes a DF of 100 to credit the scrubbing effect of the water overlying the spent fuel and the retention of the building.

## 3.3 Configuration 3 - All Fuel Stored in an Independent Spent Fuel Storage Installation (ISFSI)

As discussed in Section 2, after a sufficient decay period, long-term spent fuel storage outside the spent fuel pool becomes a possibility. The decision to apply for a Part 72 license and to transfer all fuel to an onsite ISFSI is a licensee decision that is based, in part, on such plant-specific factors as the timing and method of plant decommissioning,* the preexistence of a licensed ISFSI, and the anticipated start of fuel shipments to a DOE facility. This section discusses the supporting assumptions for Configuration 3 that differ from the previous spent fuel storage configurations.

### 3.3.1 Accident Initiator and Timing

The Configuration 3 accident initiator** is assumed to be a tornado driven missile that pierces one cask of the ISFSI. An initiator frequency is developed, for the purposes of the offsite liability discussion in Appendix B. The Electric Power Research Institute document, EPRI NP 3365, "Review of Proposed Dry Storage Concepts Using PRA,"[49] developed an initiator frequency of 6E-6 events per year for the extremely severe tornado (windspeed of 567 miles/hour) that would be necessary to generate a missile that could pierce an ISFSI cask. The report conservatively assumes the probability of missile generation,*** missile strike and impact orientation are unity. In addition, the windspeed and the missile speed are considered to be equal; no slippage is considered. Therefore, the extremely severe tornado initiator frequency is also the ISFSI cask release frequency.

BNL believes there are also additional conservatisms embedded in the development of the severe tornado initiator frequency. The frequency was based on a Zion PRA[52] initiator frequency of 1E-3 tornados/mile$^2$-year for all tornados. According to Regulatory Guide 1.76,[53] the Zion plant is in tornado Region I. Tornado Region I has the most severe design conditions. It comprises over 50% of the land area of the

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*Partial DECON or SAFSTOR could allow long-term utilization of the spent fuel pool without significant impact on the facility decommissioning plan. Complete DECON would require fuel transfer to permit decommissioning of the spent fuel pool and supporting equipment.

**Current licensing documents for spent fuel casks and modular concrete vaults do not postulate any credible accident scenarios which will breach the ISFSI.[24-25]

***The vast majority of missiles do not have the rigidity, shape, or weight to pierce the ISFSI cask.

contiguous United States, or in excess of 1,560,000 square miles. Everything else being equal, we would expect to see an average of:

1,560,000 miles$^2$ x 6E-6 extremely severe tornados/mile$^2$ - year = 9 extremely severe tornado events per year.

Although windspeeds have been estimated that are in excess of 450 mph,[54] to the best of our knowledge, there has never been a tornado of the magnitude that would be necessary to fail an ISFSI cask.

The equation used in the EPRI report to estimate the annual probability of exceeding a velocity V at a site is:

$$P(V) = \lambda \left( \frac{V_d}{V} \right)^{\frac{1}{k}} R'(V)$$

where $\lambda$ = local mean rate of occurrence of tornadoes per square mile per year.
$V_d$ = gale velocity
k = 0.5 to 1.6 a parameter value depending on a given storm, and conservatively recommended as 1.6 until such time as additional data becomes available

R′ (V) = 17.4 exp (-0.014v for V≥290 mph,

(As developed in Reference 54.)

The factor R′(V) is an approximation (based on tornado data) that accounts for the relative frequency of different tornado events, with their respective peak velocities and correlated path dimensions. Since a tornado of the magnitude of the ISFSI initiator exceeds the information that was used to develop (R′(V), the use of this equation is suspect.

On the bases of the frequency discussion, we believe that the initiator frequency of this extremely severe tornado is overstated. In our judgement, the frequency should be at least 2 orders of magnitude less.*

**Table 3.3 Configuration 3 Release Fractions**

| NG | I | Cs | Te | Sr | Rw | La | Ce | Ba |
|---|---|---|---|---|---|---|---|---|
| 0.40 | 1.5E-5 | 2.25E-5 | 1.5E-5 | 1.5E-5 | 1.5E-5 | 1.5E-5 | 1.5E-5 | 1.5E-5 |

*This judgement is supported by NUREG/CR-4461, "Tornado Climatology of the Contiguous United States,"[55] which identifies the windspeed of $10^{-7}$ probability of tornado strike for all of the U.S. to be significantly less than that required to pierce an ISFSI cask. The staff has referenced NUREG/CR-4461 in the advanced reactors evaluations and is using the same to develop new guidance with less maximum windspeeds for tornado design criteria.

With regard to accident timing, although 10CFR Part 72 allows a minimum in-pool decay time of one year the current vendor requirements and license submittals specify five-to-ten year minimum decay times.[24-25] This study assumed accident initiation at five years after final shutdown.

### 3.3.2 Exclusion Area and Meteorology

In accordance with 10CFR72.106, this study assumes the distance from the ISFSI to the exclusion area is 100 meters. The onsite weather modeling assumes "A" stability weather with a high wind speed (30 meters/second), approximating the rapid dilution associated with a tornado to develop an estimated dose at the exclusion boundary. The offsite dose model uses the MACCS code. As discussed in Section 4, the use of MACCS under these conditions adds additional uncertainty, but the authors believe the results obtained beyond the exclusion boundary are a conservative approximation.

### 3.3.3 Accident Inventory and Source Term

The storage capacity varies for each ISFSI type. A metal or concrete storage cask can accommodate 28 PWR or 56 BWR fuel assemblies. Each NUHOMS unit has a slightly smaller design capacity of 24 PWR or 52 BWR assemblies.[26] This study utilized the higher capacity cask inventories and further assumed the high burnup of the previous configurations, 60,000 (PWR) and 40,000 (BWR) MWD/MTU.* The DOE spent fuel database[15] was again used to assemble the quantities of radionuclides for input into the MACCS code.

Licensed ISFSIs are substantial engineered enclosures. The catastrophic failure of the current designs is not believed to be credible. Any damage to the ISFSI and the contained fuel is expected to be limited. Therefore, the accident inventory assumes that all of the fuel rods in one assembly are breached.

The best estimate release fractions for Configuration 3 were developed by a peer group.[39] The group reviewed published information,[21,40-42] and considered the effect of high burnup on the particulate release fractions to the cask. Since the ISFSI design pressure is slightly above atmospheric ($\sim$0.4 bar), there could be a slight driving force to the environment if the cask integrity is compromised. A bounding calculation was performed to estimate the fission product retention. Assuming isentropic expansion of the gas within the ISFSI and an environmental pressure associated with a tornado, a decontamination factor (DF) of about 2 was obtained. The Configuration 3 release fractions are presented in Table 3.3.

## 3.4 Configuration 4 - All Fuel Removed from the Site

In the future, when a DOE MRS (or a high level waste repository) becomes operational, the option of offsite storage (or disposal) of spent fuel will become available. At that time, the DOE will begin accepting spent fuel shipments with a minimum of five years decay.[27] In order to envelope future plant

*Although presently limited to a maximum burnup level of 40,000 MWD/MTU it is anticipated that future ISFSI storage concepts will be licensed for high burnup fuel.

shutdowns when the offsite shipment of fuel can be accommodated, this configuration assumed a five year onsite decay prior to the start of Configuration 4.

Publicly available literature[3-4] was reviewed to identify potential accidents that could occur during the decommissioning of nuclear power plants.

After all the spent fuel has been removed from the site, the estimated inventory that remains, although considerable, is primarily attributable to activated reactor components and structural materials. There are no credible accident sequences that can mobilize a significant portion of this activity. As a result, the potential accidents that could occur during the decommissioning of a nuclear power reactor in Configuration 4 have negligible offsite and onsite consequences. In order to develop onsite property damage insurance recommendations for Configuration 4, a rupture of the borated water storage tank is postulated.[43,44] To support the offsite liability insurance discussions of Appendix B a tank rupture initiator was developed assuming a seismic induced failure. The initiator frequency is approximately 2E-7 events per year based on a tank fragility from Reference 50 and a seismicity curve representative of the eastern United States from Reference 51. Although the health effects are negligible, the cleanup costs are significant.

# 4 RESULTS OF THE CONSEQUENCE ANALYSES

The MELCOR Accident Consequence Code System, MACCS[16-17] was used in this study to model offsite consequences. The principal phenomena considered in MACCS are atmospheric transport, mitigative actions based on dose projection, dose accumulation by a number of pathways (including food and water ingestion), early and latent health effects, and economic costs.

The prediction of onsite consequences (occupational doses) has traditionally been estimated through deterministic calculation of dose rate(s), dose(s) and contamination level(s), generally of a scoping or bounding character. Typical of these methods, was the guidance provided by Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."[28] A typical application of this method was documented in NUREG/CR-5771.[29]

In this study, a variety of deterministic methods were applied. These included the standard method as outlined in relevant Reg. Guides, and/or alternate methods, such as the Ramsdell model,[30] for estimating the concentration of material entrained in the building wake. The methods are important for predicting on-site consequences, a region generally not modelled adequately by the MACCS code.

## 4.1 Configuration 1 - Results

A series of MACCS code calculations were performed to quantify the postulated accidents cases for the Configuration 1 conditions described in Section 3.1. For each accident, Cases 1 through 4, and each generic reactor type, two calculations were performed: one using the set of high release fractions (H) and a second employing the set of low release fractions (L). The latter generally included a DF of 10 for particulates to reflect potential for retention of activity in structures. The results are tabulated in Tables 4.1 and 4.2.

A case by case comparison of the results for Configuration 1 indicates that the generic PWR and BWR results are very similar. Generally, the results are within 20 percent of one another, although in a few comparisons the differences may be somewhat larger. This similarity would be expected on the basis of *identical* site assumptions, weather conditions, interdiction criteria, and source term fractional releases adopted for both reactor evaluations. PWR inventories were generally larger than corresponding BWR inventories. The higher PWR consequences were attributable to the assumed higher burnup, the inclusion of the last normal refueling discharge in cases where the last core discharge was considered, and the relatively larger PWR pool size in the cases that considered full pool involvement.

**Table 4.1 Mean PWR Consequences**

| Accident | Inventory | Distance (miles) | Prompt Fatalities | Societal Dose (person-rem x10⁶) | Latent Fatalities | Condemned Land (sq. miles) | Total Cost ($x10⁹)** |
|---|---|---|---|---|---|---|---|
| Case 1H | full pool | 0-50 | 70 | 74 | 31,300 | 467 | 287 |
| | | 0-500 | 95 | 339 | 143,000 | 2790 | 566 |
| Case 1L | full pool | 0-50 | 1.2 | 62 | 25,300 | 297 | 100 |
| | | 0-500 | 1.2 | 130 | 53,800 | 869 | 117 |
| Case 2H | last core* | 0-50 | 29 | 81 | 33,200 | 286 | 186 |
| | | 0-500 | 33 | 226 | 94,600 | 776 | 274 |
| Case 2L | last core* | 0-50 | 0.3 | 42 | 16,800 | 156 | 56 |
| | | 0-500 | 0.3 | 70 | 28,800 | 188 | 59 |
| Case 3H | 50% pool | 0-50 | 0 | 32 | 13,200 | 25 | 25 |
| | | 0-500 | 0 | 48 | 20,400 | 25 | 25 |
| Case 3L | 50% pool | 0-50 | 0 | 6 | 2,400 | 2 | 1.1 |
| | | 0-500 | 0 | 8 | 3,400 | 2 | 1.1 |
| Case 4H | last core* | 0-50 | 0 | 24 | 10,100 | 15 | 15 |
| | | 0-500 | 0 | 36 | 15,400 | 15 | 15 |
| Case 4L | last core* | 0-50 | 0 | 4 | 1,500 | 1 | 0.8 |
| | | 0-500 | 0 | 5 | 2,300 | 1 | 0.8 |

* The "last core" also includes the last normal refueling discharge.
** excludes health effects

A limited comparison can be made of the results obtained in this effort with those of previous investigations. The consequence estimates obtained here are generally higher. For example, the societal dose commitment (0 to 50 miles) for the worst case accident (fire, full pool involvement, high release fractions) reported by Sailor[7] was 2.6 million person-rem; Jo[8] reported 25.6 million person-rem; while in the present work 75.3 million person-rem (BWR) was obtained. As discussed in Appendix A, these early efforts used identical inventory and source term assumptions. The differences observed were primarily due to the population assumptions. The average population density (0-50 miles which includes the large city) used herein was about 1800 persons per square mile. This would support an approximate increase
of a factor of two over the dose reported by Jo. The second major reason the consequences are greater is the radionuclide inventory used here. The assumptions made for reactor power, end of plant life fuel burnup and fuel pool capacity, resulted in an inventory which has substantially higher quantities of the long lived radionuclides than previous studies. For example, the total BWR pool inventory of Cs-137 was about a factor of 3 greater than developed by Sailor for the Millstone plant. Thus, the limited comparisons would indicate that the consequences determined in this study were generally higher than the former studies. The consequences are consistent with earlier work, when gross differences in the underlying assumptions are taken into account.

**Table 4.2 Mean BWR Consequences**

| Accident | Inventory | Distance (miles) | Prompt Fatalities | Societal Dose (person-rem $x10^6$ | Latent Fatalities | Condemned Land (sq. miles) | Total Cost $\$x10^9$** |
|---|---|---|---|---|---|---|---|
| Case 1H | full pool | 0-50 | 74 | 75 | 31,900 | 456 | 280 |
| | | 0-500 | 101 | 327 | 138,000 | 2170 | 546 |
| Case 1L | full pool | 0-50 | 1.3 | 58 | 23,600 | 286 | 97 |
| | | 0-500 | 1.3 | 120 | 49,800 | 784 | 113 |
| Case 2H | last core | 0-50 | 24 | 81 | 33,000 | 262 | 167 |
| | | 0-500 | 26 | 207 | 86,400 | 521 | 234 |
| Case 2L | last core | 0-50 | 0.2 | 38 | 15,300 | 140 | 48 |
| | | 0-500 | 0.2 | 62 | 25,700 | 159 | 51 |
| Case 3H | 50% pool | 0-50 | 0 | 29 | 12,200 | 23 | 23 |
| | | 0-500 | 0 | 45 | 18,900 | 23 | 23 |
| Case 3L | 50% pool | 0-50 | 0 | 5 | 2,100 | 2 | 1.0 |
| | | 0-500 | 0 | 7 | 3,000 | 2 | 1.0 |
| Case 4H | last core | 0-50 | 0 | 20 | 8,300 | 13 | 12 |
| | | 0-500 | 0 | 30 | 12,700 | 13 | 12 |
| Case 4L | last core | 0-50 | 0 | 3 | 1,300 | 1 | 0.7 |
| | | 0-500 | 0 | 4 | 1,900 | 1 | 0.7 |

** excludes health effects

The total costs of fuel pool accidents observed in this study were found to rise more sharply than the societal dose. This reflects the tradeoffs of protective (interdiction and relocation) actions. These actions are, of course, intended to limit public exposure to the released radioactivity, but at the increased cost of primarily population dependent interdiction and relocation expenses. Again the major obvious factors, which will drive costs up in comparison to earlier studies, are the larger population at risk and the larger inventory of material considered in this study. This observation is supported by a comparison of the condemned land. Comparing Case 1H in Table 4.1 or 4.2 with case 1A of Table A.2, it can be seen that the condemned area has doubled. Although, Table A.2 identifies this as interdicted area, which might be subject to a different interpretation given the usage of this term by the MACCS code, the text of the Sailor study clearly stated "... interdicted area (the area with such a high level of radiation that it is assumed that it cannot ever be decontaminated)." Condemned land is defined as farmland permanently removed from production, as such it does not account for the population affected area. However, the condemned area for case 1H in the present study clearly indicates a more extensive contamination of all lands when compared to the former study. This increase translates into increased costs.

**Table 4.3 PWR Core Melt Accident Results**

| Accident | Inventory | Distance (miles) | Prompt Fatalities | Societal Dose (person-rem x10⁶) | Latent Fatalities | Condemned Land (sq. miles) | Total Cost ($x10⁶) |
|---|---|---|---|---|---|---|---|
| RZ1 with evacuation | 3800 Mwt core | 0-500 | 88 | 70.* | 35,000 | 2000 | NR |
| RZ1 no evacuation | 3800 Mwt core | 0-500 | 160 | 220. | 110,000 | 2000 | NR |
| * Doses that were not reported, have been estimated from the number of latent fatalities and the BEIR-V recommended risk coefficient of 5.0E-4 fatalities per person-rem. | | | | | | | |

(Reproduced from Reference 14)

For perspective, it is interesting to provide some comparison to core melt accidents. A major core melt accident (RZ1, large early release) was selected from the results reported in Reference 14. This study employed many of the assumptions, i.e., population distribution and weather conditions, that were employed in the present analysis, thus allowing for reasonable comparison. The core melt accident source term was 100% of the noble gases, 27% of the iodine group, 21% of the cesium group, 10% of the tellurium group, 12% of the barium and strontium groups, 0.52% of the ruthenium group, 0.2% of the lanthanum group and 0.6% of the cerium group. Table 4.3 summarizes the reported results.

The core melt accident results are provided for two emergency protective actions: one in which a representative evacuation was modelled along with long term protective actions; and a no evacuation, no long term protective action case. The later case, while unrealistic, provides a very conservative bounding estimate of the consequences. A case with protective actions identical to this study was not reported. However, the results of such an analysis would have provided results intermediate to those reported (with the exception to condemned land which is not affected by emergency response). Comparison with the results shown in Tables 4.1 and 4.2 clearly indicates that for worst case assumptions, i.e., full pool involvement and large source term, the postulated Configuration 1 spent fuel pool accident may have *comparable* consequences to a major core melt accident.

Previous studies have elected to quantify the risks and costs of fuel pool accidents using either Case 1 or Case 2 results. In their final analysis, Sailor, et al.,[7] chose the last refueling offload/maximum source term accident results. In Jo, et al.,[8] a worst case (full pool/maximum source term accident) and a best estimate case (last refueling/maximum source term accident) were explored. For the present evaluation, BNL recommends that the estimated consequences for case 2L be used. This case assumes that the accident is limited to the last full core discharge (plus the last normal refueling discharge in the case of a PWR) and the lower release fractions, that reflect some credit for fission product retention.

This recommendation has been made for the following reasons. As discussed in NUREG/CR-4982, there is a large degree of uncertainty associated with the fire propagation throughout the entire pool. Additionally, mitigative options such as rack modifications,[5-6] (i.e., increased hole size) and fuel

management practices (including checkerboarding of fresh assemblies and the use of regions in the SFP) are all possible. Thus, it is possible to reduce the likelihood of propagation into the older assemblies. Regarding the lower fractional releases in the recommended case, BNL considered the implications of the accident that occurred at the Chernobyl Unit-4 power plant in the Ukraine.[31] Although Chernobyl is clearly not an analog of the accidents treated in this section, several similarities exist which have relevance to the fuel pool accident. These include oxidation of the clad, failed reactor structure and the availability of air. (There are of course many dissimilarities, such as the burning of the graphite moderator which provided additional heating and the expulsion of fuel fragments to the environment during the violent steam explosion.) Nonetheless, it is difficult to envision that the spent fuel pool accident(s) could result in much greater release. The estimated Chernobyl release, as a fraction of core inventory, was 1.0 of the noble gases, 2.0E-1 of the iodine, ~1.3E-1 of the cesium and tellurium, 4.0E-2 of the strontium, 5.6E-2 of the barium, and approximately 3.0E-2 of the ruthenium, cerium and lanthanum group nuclides.

A comparison with the source terms in Table 3.2, shows better agreement for the noble gas (NG), I and Cs groups with the low (Case 2) release source term. In contrast, the Chernobyl releases for Te and the nonvolatiles greatly exceed any of the releases shown. There are two justifications for the lower Te and nonvolatile group releases used in this study. In the case of Te, the formation of an intermetallic compound with Zr in the clad is known to suppress Te release until the clad is completely oxidized. At Chernobyl, complete oxidation of the clad probably occurred in the rubble bed that the reactor became. In the spent fuel pool accident, Sailor et al. believed that cladding would melt prior to complete oxidation, relocate and be quenched on the floor of the pool. The cladding material would thus retain Te.

## 4.2 Configuration 2 - Results

The offsite consequences for Configuration 2, "Cold Fuel in the Spent Fuel Pool," were modeled with the MACCS code using the input assumptions of Section 3.2. The deterministic treatment outlined in Reg. Guide 1.25 was not pursued because it provided a limited description of the consequences.*

The estimated offsite consequences for each reactor type and assumed environmental release is shown in Table 4.4.

As expected, these results indicate a far lower level of offsite consequences than the Configuration 1 cases. The much lower inventory is the obvious reason for the low level of predicted accident consequences. In no case is prompt fatalities indicated. Societal doses are very much lower than those developed for Configuration 1 accidents. These low doses are reflected in the low numbers of latent fatalities estimated. For either reactor type a very small area of farmland is predicted to be permanently condemned, only when the high gap release fractions (worst case assumptions) are employed. These lands are well within 10 miles of the plant. When the low gap release fraction (central estimate) was

*The Reg. Guide 1.25 methodology is limited to noble gases and iodine. The extension of this methodology to address the small fraction of particulates postulated for Configuration 2 is beyond the scope of this program.

employed, the condemnation of land was not predicted. The estimated total off site cost, excluding health costs, range from 28 million dollars to negligible, dependent on reactor type and release assumptions. These costs are very much lower than the Configuration 1 accident.

**Table 4.4 Mean Offsite Consequences - Configuration 2**

| Generic Plant Type | Release Characterization | Distance (miles) | Prompt Fatalities | Societal Dose (person-rem) | Latent Fatalities | Condemned Land (sq. miles) | Total Cost** $x10$^6$ |
|---|---|---|---|---|---|---|---|
| PWR | High gap | 0-50<br>0-500 | 0<br>0 | 2.E+5<br>3.E+5 | 100<br>134 | 0.03* | 28 |
| PWR | Low gap | 0-50<br>0-500 | 0<br>0 | 3.E+3<br>4.E+3 | 1<br>2 | 0.0 | neg. |
| BWR | High gap | 0-50<br>0-500 | 0<br>0 | 7.E+4<br>9.E+4 | 31<br>40 | 0.002* | 6 |
| BWR | Low gap | 0-50<br>0-500 | 0<br>0 | 8.E+2<br>1.E+3 | 0.4<br>0.5 | 0.0<br>0.0 | neg. |

* Indicates within 10 miles of plant, ** excludes health effects, "neg" denotes negligible

To estimate the dose at the site boundary (0.4 miles beyond the point of release) the MACCS calculations were repeated, since centerline dose was not predicted for the "relocation only" emergency response. The code requires an evacuation model to calculate centerline dose. To maximize time in the plume. BNL chose a ten hour delay to start the evacuation. Thus, individuals near the site boundary were exposed for ten hours to the release, then evacuated. The lifetime whole body effective dose equivalent for this exposure was calculated. Both the high and low source terms assumed for Configuration 2 were evaluated. As calculated by the MACCS code, these doses included exposure from all direct pathways.

In the mean, the doses at the site boundary were estimated to be 930 and 0.9 mrem, for the high and low PWR Configuration 2 release assumptions. The BWR doses were estimated to be about a factor of 4 lower.

For the purpose of regulatory requirement analysis, it is recommended that the consequences developed with low fractional releases be employed. The consequences estimated with the high gap releases should be viewed as an upper limit, as no credit is taken for retention in the pool or in the undamaged housing structure. Clearly, some level of fission product retention in the pool and in the structure is to be expected. The low fractional releases therefore would appear to provide a more reasonable estimate of the actual releases that could occur.

**Configuration 2 - Onsite Consequences**

Onsite dose assessments were performed with the Ramsdell model[30] and the model provided in Reg. Guide 1.145.[32] These deterministic analyses, which take into account the entrainment of the release into a building wake, were performed for two polar weather conditions to provide an indication of the range of anticipated dose(s). Descriptions of these dispersion/dose models are provided in Reference 30. For the Ramsdell model, unstable A and stable G weather conditions were evaluated at a 1 meter/sec wind speed. For the Reg. Guide 1.145 model, Class A and F weather were evaluated. The release was assumed to occur at a height of 10 meters and the reactor structure had an effective area of 1500 square meters which enters into the description

**Table 4.5 Configuration 2 Estimates of the Committed 50 Year Dose to a Worker**

| Model | Weather Stability | Committed Dose (rem) | |
|---|---|---|---|
| | | PWR | BWR |
| Ramsdell | A | 0.88 | 0.24 |
| | G | 1.23 | 0.33 |
| Reg. Guide 1.145 | A | 0.60 | 0.16 |
| | F | 4.24 | 1.14 |

of the building wake. The integral 50 year effective whole body dose commitment from cloudshine and inhalation were estimated 100 meters downwind of the release. The necessary dose conversion factors were taken from the MACCS code DOSDATA file.[16] These calculations conservatively assumed an individual is immersed in the release plume for the entire 2 hour duration of the release.

Table 4.5 provides the estimated on site ("parking lot") dose assessment. Only the lower release for each generic reactor type was evaluated.

The range of dose is dependent on both the assumed weather conditions at the time of release and the model that was employed to arrive at the result. In all cases, the estimated doses for the single assembly fuel handling accident are relatively low.

Since the Ramsdell model has been developed more recently than the regulatory guidance and since it has been based on the results of experimentation, the authors were inclined to place more confidence in its estimates. Thus assuming stable weather condition G at the time of release for a degree of conservatism, the onsite worker dose from the postulated fuel handling accident were estimated at 1.2 and 0.3 rem, PWR and BWR, respectively.

The cleanup and decontamination costs for the Configuration 2 fuel handling accident were estimated using the cost estimates provided in a study performed by Pacific Northwest Laboratories (PNL).[34] Three reactor accident regimes were considered in the PNL study. The least severe of these regimes, assumed

that the accident involved a 10% cladding failure, no fuel melting, moderate contamination of structures and no significant damage to the physical plant. While the extent of assumed fuel damage was greater than the single assembly fuel handling accident, several similarities are observed. The cleanup and decontamination of the plant structure(s) to bring the plant the site to a safe condition will require damaged fuel removal, water cleanup, and surface decontamination of walls, floors, etc. Since a release of fuel fines for a mechanical disruption of the fuel cladding is postulated, and complete retention in the pool coolant is not assured, potential fission product contamination of the interior of the structure housing the spent fuel pool must be assumed. As such, the estimate developed by PNL provides a basis for estimating the cleanup cost of a fuel pool accident. The costs were $98 and 72 million (1981$) for BWR and PWR plants, respectively. If we assume that the extent of contamination and complexity of cleanup and decontamination are proportional to material at risk in the respective accidents and the cleanup cost escalates at 5% per year, the BWR and PWR costs for a fuel handling accident are $2.7. and 7.8 million dollars, respectively. Since these costs may not be totally elastic, a contingency factor of three has been added. This places the total onsite cost at approximately $9 to 24 million dollars. These costs are relatively small and further quantification is not believed to be necessary for this analysis.

## 4.3 Configuration 3 - Results

Offsite consequences were again modelled with the MACCS code. The identical set of assumptions that were employed in the Configuration 1 and 2 analyses were used for Configuration 3 with the following exceptions: the exclusion boundary was 100 meters; the release height was 1 meter; and the height and effective width of the ISFSI were 2 and 6 meters, respectively. The appropriate Configuration 3 inventories and source terms were used. The use of the MACCS code, or for that matter any Gaussian dispersion model, at a distance of 100 meters is debatable. It is generally agreed that the experimentally determined dispersion parameters, and more importantly, the analytical expressions used within the MACCS code to summarize this data, provided a better picture of plume behavior at a distance greater than several hundred meters. Thus, the estimated results of the MACCS code close to the point of release are subject to an additional degree of uncertainty, whereas results beyond several hundred meters are not. However, this limitation is minor in comparison to the limitation discussed below.

The standard treatment of estimating offsite consequences with the MACCS code, and in particular sampling representative weather conditions, is in conflict with the assumed accident scenario. The accident was assumed to be initiated by a tornado driven missile with resultant very rapid release of material. The weather conditions at the time of release are therefore more accurately described as high turbulence with very high velocity winds. Accurate treatment of these conditions is beyond the capabilities of the MACCS code. However, the results obtained with the code executed in the typical fashion of accident analysis, should provide a conservative estimate of the accident consequences. (It can be stated that the anticipated dispersion occurring in the wake of a tornado would be much greater than that predicted for practically all other weather conditions).

The estimated offsite consequences for each type of reactor fuel is presented in Table 4.6.

The offsite consequence estimates provided in Table 4.6 are qualitatively comparable to those obtained for Configuration 2, and low in comparison to Configuration 1.

To obtain an estimate of the dose at the site boundary (for Configuration 3 the site boundary was placed at 100 meters beyond the point of release), the MACCS calculations were not repeated as was the case for Configuration 2. The results of the Reg. Guide 1.145 treatment,[32] which were intended to assess worker exposures, also serve as a reasonable estimate of the dose at the site boundary, since the ISFSIs were located 100 meters from the exclusion boundary in this study. The 50 year committed doses are 472 millirem for the PWR and 82 millirem for the BWR. The difference in estimated committed doses is primarily attributable to the greater nuclide inventory and the higher burnup associated with the PWR assembly.

**Table 4.6 Mean Offsite Consequences - Configuration 3**

| Generic Plant Type | Distance (miles) | Prompt Fatalities | Societal Dose (person-rem) | Latent Fatalities | Condemned Land (sq. miles) | Total Cost ($x10$^6$)* |
|---|---|---|---|---|---|---|
| PWR | 0-50 | 0 | 5.9E+2 | 1.8E-1 | 0.002 | neg. |
| | 0-500 | 0 | 6.9E+2 | 2.2E-1 | | |
| BWR | 0-50 | 0 | 1.2E+2 | 4.2E-2 | 0.0 | neg. |
| | 0-500 | 0 | 1.5E+2 | 5.1E-2 | | |

"neg" denotes negligible, * excludes health effects

Onsite costs for Configuration 3 are estimated to be the sum of the replacement cost of the damaged cask and of the removal and disposal cost of contaminated soil. The cost of an ISFSI cask is $0.75 to 1 million dollars. The onsite area that is contaminated is estimated to be 0.002 square miles. Assuming the affected soil is removed to a depth of 3 inches and a disposal cost of $320.00 per cubic foot, the soil cleanup costs are approximately 5 million dollars. The total estimated costs are about 12 million dollars, including a contingency factor of about two.

## 4.4 Configuration 4 - Results

After all the spent fuel has been removed from the site, the radionuclide inventory that remains, although considerable, primarily consists of activated reactor components and structural materials. There are no credible accidents that can mobilize a significant portion of this activity. Previous studies[3,4] have estimated that routine and postulated accident releases to the environment were in the range of $\mu$Ci to 10 mCi. Releases of this magnitude are also expected to result in negligible onsite accident worker doses and negligible onsite contamination.

For the purpose of estimating onsite accident cost one could consider an accident at a power plant similar to the postulated borated water tank rupture accident that was discussed in the Rancho Seco exemption

request.[35] This scenario postulated that the most severe accident was the postulated rupture of the borated water storage tank (BWST) which could release about 450,000 gallons of slightly radioactive water onto the plant grounds. The level of released activity was small, but it was assumed that a cleanup of the grounds would be required. The cost of cleanup is driven by the volume of liquid and not directly by the level of activity in the water. This is illustrated by Tables 4.7 and 4.8 which present the expected concentration of radioisotopes in the BWST. Table 4.7 presents the expected level of short-lived radioisotopes, while Table 4.8 provides the level of long lived radioisotopes at selected times after shutdown. Most of the radioisotopes listed in Table 4.7 decay to nothing within 120 days, and virtually all are gone after 1 year.

At Rancho Seco, the BWST has a capacity of 450,000 gals. The activity of this water is extremely low, and after 5 years is primarily due to tritium with an activity of 5000 curies, (a soft beta emitter) and approximately 60 mCi of Cs-137. This amount of radioactivity is generally considered to be a trace contamination; all the shorter half-lived nuclides, shown on Table 4.8, have decayed away. The cleanup estimate developed by the Sacramento Municipal Utility District (SMUD) for the Rancho Seco plant primarily consisted of the removal and disposal of 18 inches of gravel and two feet of the underlying soil in the vicinity of the BWST. This would result in the disposal of about 150,000 $ft^3$ of soil. SMUD assumed a 1991 waste disposal cost of $150.00 per cubic foot. Waste transportation costs were neglected.

BNL modified the Rancho Seco plant specific estimate to make it more generic by using the 1995 disposal cost of $320.00/$ft^3$ for the Barnwell facility.[36] This results in a cleanup cost of about $54 million.*

However, it is likely that much of this contaminated water would migrate toward the water table and not be captured by the mechanical removal of the surface soil. The contaminated water could reach the water table below the site and result in tritium levels in excess of the maximum concentration limit for drinking water. BNL has calculated that in the time it takes the plume to reach the site boundary, radioactive decay and dispersion could be expected to reduce the tritium concentration below the maximum concentration limit for drinking water, thus it is assumed no treatment would be required.

In order to encompass the cost of onsite groundwater characterization, groundwater monitoring and sample testing over approximately 60 years, the waste disposal estimate of $54 million has been multiplied by a factor of ~2 to $110 million.

*Consisting of removal, disposal and restoration costs. Waste transportation costs were neglected.

**Table 4.7 Activity of the Short-Lived Isotopes in the Boric Acid Concentration Tanks**

| Isotope | Concentration ($\mu$Ci/ml) | Max. Activity ($\mu$Ci in 450,000 gal) |
|---|---|---|
| I-131 | 2.45E-08 | 50.11965 |
| I-132 | 0 | 0 |
| I-135 | 3.36E-09 | 6.873552 |
| I-136 | 1.09E-11 | 0.22298 |
| Cs-136 | 6.23E-07 | 1274.471 |
| Mo-99 | 1.90E-07 | 388.683 |
| Y-90 | 9.20E-09 | 18.82044 |
| Kr-85m | 1.24E-13 | 0.000254 |
| Kr-88 | 0 | 0 |
| Xe-131m | 9.52E-09 | 19.47506 |
| Xe-133 | 8.57E-07 | 1753.165 |
| Xe-133m | 5.88E-09 | 12.02872 |
| Xe-135 | 1.80E-10 | 0.368226 |
| Y-91 | 5.34E-08 | 109.2404 |

**Table 4.8 Activity of the Long-Lived Isotopes in the Boric Acid Concentrate Tanks**

| Isotope | Concentration ($\mu$Ci/ml) | Activity (Ci) in 450,000 gal. | | | | | |
|---|---|---|---|---|---|---|---|
| | | Initial Activity | @120D | @1 yr | @2 yr | @5 yr | @10 yr |
| H-3 | 2.5 | 5110 | 5020 | 4830 | 4570 | 3860 | 2910 |
| Cs-137 | 0.00003 | .0610 | .0605 | .0596 | .0582 | .0543 | .0484 |
| Kr-85+ | 3.30E-08 | 6.7(-4)* | 6.6(-4) | 6.3(-4) | 5.9(-4) | 4.8(-4) | 3.5(-4) |

+Assumed release to atmosphere at time of spill
*6.7(-4) - $6.7 \times 10^{-4}$

# 5 REGULATORY ASSESSMENT SUMMARY

The preceding sections of this report have provided an overview of the processes that are likely to occur when a nuclear power plant permanently ceases operation. The primary focus of this study has been the storage alteratives for the spent fuel. Section 4 examined multiple cases for each spent fuel configuration. A "best estimate" case/consequence analysis was presented for each spent fuel storage configuration including: societal dose, latent fatalities, the amount of condemned land, and the estimated cost of the postulated accident.

After a plant is permanently shutdown, awaiting or in the decommissioning process, certain operating based regulations (or technical issues) may no longer be applicable. The purpose of this section is to present the results of this regulatory assessment.

A list of candidate regulations was identified from a screening of 10CFR Parts 0-199.[37] Each of these technical issues was subjected to a detailed review which included federal register notices, SECYs, NRC policy statements, regulatory guides, standard review plans, NUREGs, NUREG/CRs,
\etc, to develop an understanding of the regulatory
bases. The continued applicability of each technical issue was assessed within the context of each spent fuel storage configuration, the results of the consequence analyses, as well as the expected plant status.

With the possible exception of Part 171, "Annual Fees for Licensees," each regulation is ultimately focussed on the protection of public health and safety. However, a particular regulation may not be applicable to a permanently shutdown plant in general, or a specific spent fuel storage configuration. For example, an exemption from the containment leakage testing requirements of 10CFR50.54(o) for a permanently defueled plant will not impact public health and safety as the plant risk is primarily associated with the spent fuel that is now stored in the spent fuel pool outside the primary containment.

The results of the regulatory assessment are presented in Table 5.1. The detailed recommendations, including regulatory background, specific cites, and regulatory assessment are included as Appendix B to this report.

**Table 5.1 Assessment of Continued Regulatory Applicability for Permanently Shutdown Nuclear Power Reactors (Summary)**

| | | Regulatory Applicability[2-5, 30] | | | | |
|---|---|---|---|---|---|---|
| | | Configuration | | | | |
| Technical Issue | 10CFR Reference[1] | 1 | 2 | 3[6-8] | 4 | Notes |
| Fitness for Duty | Part 26<br>55.53(j), (k)<br>72.194 | P<br>F | N<br>N | N<br>N<br>F | N<br>N | 9,10 |
| Technical Specifications | 50.36, .36a, .36b<br>72.26, 72.44 | P | P | P<br>F | P | 11 |
| Combustible Gas Control | 50.44 | N | N | N | N | |
| ECCS Acceptance Criteria | 50.46 | N | N | N | N | |
| Emergency Planning and Preparedness | 50.47, .54(q),(t) App. E<br>72.32 | F | P | P<br>F | P | 12,13 |
| Fire Protection | 50.48, App. R<br>72.122 | P | P | P<br>F | P | 15 |
| Environmental Qualification | 50.49 | N | N | N | N | |
| QA Program | 50.54(a), App. B<br>Part 72, Subpart G | P | P | P<br>F | P | 16 |
| Operator Requalification Program | 50.54(i), 55.45, 55.59<br>72.44(b) Part 72, Subpart I | P | P | N<br>F | N | 17 |
| Operator Staffing Requirements | 50.54(k),<br>50.54(m) | N<br>F | N<br>P | N<br>N | N<br>N | 18 |
| Containment Leakage Testing | 50.54(o), App. J | N | N | N | N | |
| Security Plan | 50.54(p), 70.32, Part 73<br>Part 73 App. B and C<br>72.44(e) Part 72 Subpart H<br>Part 73 | P | P | N<br><br>F | N | 19 |
| Onsite Property Damage Insurance | 50.54(w)<br>Part 72 | F | P | P<br>* | P | |
| Inservice Inspection Requirements | 50.55a(g) | P | P | N | N | 21 |
| Fracture Prevention Measures | 50.60, .61, Apps. G and H | N | N | N | N | |
| ATWS Requirements | 50.62 | N | N | N | N | |

| | | Regulatory Applicability[2-5, 30] | | | | |
|---|---|---|---|---|---|---|
| | | Configuration | | | | |
| Technical Issue | 10CFR Reference[1] | 1 | 2 | 3[6-8] | 4 | Notes |
| Fitness for Duty | Part 26<br>55.53(j), (k)<br>72.194 | P<br>F | N<br>N | N<br>N<br>F | N<br>N | 9,10 |
| Loss of all AC Power | 50.63<br>72.122(k) | N | N | N<br>F | N | 22<br>23 |
| Maintenance Effectiveness | 50.65 POL before 7/10/96<br>POL after 7/10/96 | N<br>P | N<br>P | N<br>N | N<br>N | 24 |
| Periodic FSAR Update Requirement | 50.71(e)<br>72.70 | P | P | P<br>F | PN | <br>26 |
| Training and Qualification of Nuclear Power Plant Personnel | 50.120 | P | P | N | N | |
| Material Control/Accounting of Special Nuclear Material (including US/IAEA Agreement) | 70.51, .53, 74.13(a) (Part 75)<br>72.72, .76 | F | F | N<br>F | N | 27<br>27 |
| Financial Protection Requirements | Part 140<br>Part 72 | F | P | P<br>* | P | |
| Annual Fees for Licenses | 171.15<br>171.16 | P | P | P<br>F | P | 29 |

* See discussion in Appendix B.

## NOTES TO TABLE 5.1

1. 10CFR Parts 0 to 199, revised January 1, 1995.

2. All other regulatory requirements applicable to nuclear power reactors and not listed in this table are assumed to remain in effect, unless addressed by a plant-specific exemption.

3. The spent fuel storage configurations are defined in Sections 2 and 3 of this report. Briefly:

   Configuration 1 - hot fuel in the spent fuel pool
   Configuration 2 - cold fuel in the spent fuel pool
   Configuration 3 - all fuel stored in an ISFSI
   Configuration 4 - all fuel shipped offsite

4. Configuration 1 also assumes the licensee has a Possession Only Licensee or that a confirmatory letter has been issued to prevent refueling the vessel without NRC authorization.

5. F-Regulation continues to be fully applicable for this spent fuel storage configuration.
   P-Regulation is assessed to be partially applicable for this configuration.
   N-Regulation is not considered applicable to this configuration.

6. A permanently shutdown nuclear power plant may store its fuel in an Independent Spent Fuel Storage Installation, before, during, and after the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. This necessitates two (or more) entries in Table 5.1 for Configuration 3. The first (and second, if applicable) pertains to the plant itself prior to the completion of decommissioning. The last entry examines the Part 72 requirements for the ISFSI.

7. The requirements of Configuration 3 remain applicable until all fuel has been removed from the ISFSI and shipped offsite.

8. In addition to the applicable provisions of Part 72 as noted for Configuration 3, Parts 20, 21, 71, and 73 remain applicable to the transportation of spent fuel from the ISFSI to a HLW repository or MRS.

9. Although the Part 26 requirements may no longer be appropriate for certain spent fuel storage configurations, the recordkeeping requirements of Section 26.71 are still applicable.

10. The Part 26, Fitness for Duty requirements remain applicable for Configuration 1. However, the scope of the program can be limited to those personnel with unescorted access to the fuel building.

11. The technical specification requirements are very plant specific. Plant systems and controls necessary for the continued public health and safety will vary from plant to plant. BNL

recommends a plant-specific amendment request to reduce the scope of the operating tech specs or institute defueled tech specs.

12. BNL recommends that all emergency planning and preparedness requirements remain applicable to Configuration 1, with the exception of the Emergency Response Data System (Part 50, Appendix E, VI).

13. BNL recommends site-specific calculations to establish a new smaller EPZ boundary for the plant for Configuration 2. Based on the assumption (subject to plant-specific verification that no members of the public will be exposed in excess of the EPA PAGs, BNL recommends the licensee apply for exemptions from the following Part 50 EP requirements for Configuration 2:

- The early public notification requirements of 50.47(b)(5) and Appendix E.IV.D.3.
- The periodic dissemination of emergency planning information to the public of 50.47(b)(7) and Appendix E.IV.E.8.
- Offsite emergency facilities and equipment such as the EOF, and the emergency news center (50.47(b)(8), Appendix E.IV.E.8).
- Offsite radiological assessment and monitoring capability, including field teams (50.47(b)(9)).
- Periodic offsite drills and exercises (50.47(b)(14), Appendix E.IV.F.3).
- Licensee headquarters support personnel training (50.47(b)(15), Appendix E.IV.F.b.h).

Since decommissioning accidents that do not involve spent fuel have negligible public health consequences offsite EP can also be eliminated for Configurations 3 (plant only) and 4.

14. The emergency planning requirements for ISFSIs that are not associated with an operating nuclear power plant are the subject of a final rule issued on June 22, 1995 [60FR32430].

15. Each licensee has a Fire Protection Program that, in addition to safe shutdown requirements, has training requirements, administrative procedures and controls, and detection/suppression requirements for plant areas that contain radioactive inventories with potential offsite consequences. BNL recommends deleting requirements directly related to safe shutdown capability. Further reductions in the scope of the fire protection program should be on a plant-specific basis.

16. Permanently defueled plants are expected to be able significantly to reduce the scope of their QA program without impacting public health and safety. In accordance with 50.54(a)(3), any proposed changes to the previously accepted QA program must be approved by the NRC.

17. The licensee should submit, per 10CFR50.54(i), a revised operator requalification program limited to fuel handling to reflect the defueled configuration.

18. BNL recommends that at least one licensed SRO be present or readily available on call at all times (see 50.54(m)(l)), for Configurations 1 and 2. Our concern is maintaining fuel cooling

under off normal conditions and the ability to carry out the units' emergency plan (EP), at least in its early stages.

19. In comparison to an operating unit, a permanently defueled plant has less vital equipment and a potentially smaller vital area(s). Accordingly, it is expected that these licensees will continue to apply for exemptions to reduce the scope of the plan.

20. Not used.

21. The scope of the Inservice Inspection Program can be reduced to address only those systems in the existing plan that support spent fuel storage. Some plants do not include spent fuel cooling in their program and may eliminate the Program in its entirety.

22. The intent of the Station Blackout (SBO) Rule is to maintain the risk of fuel damage due to SBO to $\sim 10^{-5}$/reactor year. Permanently shutdown plants meet the intent of 10CFR50.63. BNL recommends existing SBO plant procedures and training be revised to reflect the storage of all fuel in the spent fuel pool.

23. For Configuration 3, offsite power is required for ISFSI security and monitoring systems.

24. The Maintenance Rule does not become effective until July 10, 1996. Plants that request a POL prior to that date should not be subject to this requirement. A facility that is permanently shutdown after that date will have a program to enhance maintenance effectiveness which can be reduced to those systems that support fuel storage and handling, building ventilation and filtering, and radiation monitoring.

25. Not used.

26. ISFSIs are currently required to submit an *annual* FSAR update per 10CFR72.70.

27. The Part 70 license remains in effect until the site is released for unrestricted use. However, an exemption from the special nuclear material (SNM) control and accounting requirements of Parts 70 and 74 and the safeguards requirement of Part 75 can be issued after the SNM has been disposed of. However, please note that an ISFSI has its own requirements under Part 72.

28. Not used

29. Although the current practice is to grant full exemptions from the annual licensing fees for permanently shutdown power reactors, BNL proposes a partial exemption for future years. As the NRC experience with large power reactor decommissioning grows, a fee based on the services provided to these licensees could be applied. Alternatively, Part 171.15 fee that is equivalent to the ISFSI annual fee may be appropriate.

30. This regulatory assessment assumes an onsite, operating spent fuel pool is not necessary to satisfy the fuel retrievability requirement of 72.122(l).

# 6 SUMMARY AND CONCLUSIONS

Brookhaven National Laboratory (BNL) has undertaken a program (FIN L-2590), "Safety and Regulatory Issues Related to the Permanent Shutdown of Nuclear Power Plants Awaiting Decommissioning." This report summarizes the results of the program, which performed a regulatory assessment for generic BWR and PWR plants that have permanently ceased operation.

Previous studies have concluded that decommissioning accidents that do not involve spent fuel have negligible off-site and on-site consequences. Therefore this study focused on current and future spent fuel storage alternatives for the permanently shutdown facility. Four spent fuel storage alternatives were identified:

| | |
|---|---|
| Configuration 1 - | Hot fuel in the spent fuel pool |
| Configuration 2 - | Cold fuel in the spent fuel pool |
| Configuration 3 - | All fuel stored in an ISFSI |
| Configuration 4 - | All fuel removed from the site |

Each of these configurations was further defined to support the consequence analyses and the regulatory assessment. A set of assumptions was developed to envelope future end of life nuclear power plant shutdowns, as well as plants that have prematurely ceased operation. Thus, this study postulated: higher end of life fuel burnups than presently experienced; spent fuel pools at full capacity; and a high population density to account for future industry and population trends. In addition, this study also differs from previous efforts because the gap release source terms, used herein, are partially based on experimental results and include a small fraction of fuel fines.

**Consequence Analyses**

Several accident cases, with different inventory and release assumptions, were evaluated for each spent fuel storage configuration. Table 6.1 presents the consequences for the accident cases that were adopted for the regulatory assessment. The Configuration 1 accident postulates an event that causes the draining or boiloff of the water in the fuel pool, exposing the relatively hot spent fuel assemblies to an air environment. The most recently discharged assemblies self heat to a point where the Zircaloy oxidation becomes self sustaining, resulting in extensive clad failure and fission product release. As shown in Table 6.1, the Configuration 1 accident consequences are severe, approximating those of a core melt accident. These results are higher in comparison to previous studies. This is primarily attributable to the higher population assumption used herein. A secondary contributor is the greater radionuclide inventory. The assumptions made for reactor power, end of plant life fuel burnup and fuel pool capacity* resulted in an

*Does not impact the recommended Configuration 1 accident consequences.

inventory with substantially higher quantities of long lived radionuclides than those assumed in previous studies.*

After sufficient decay time has elapsed and the rapid oxidation phenomenon is not likely, the fuel was considered to be in Configuration 2, "Cold fuel in the spent fuel pool." The accident initiator was the drop of a single assembly, resulting in a gap release. In addition to partial releases of the noble gases and iodine (if present), small releases of the remaining nuclide groups are expected on the basis of experimentally observed releases of fuel fines. The source term for the recommended Configuration 2 accident case includes credit for the scrubbing effect of the water overlying the fuel.

As shown in Table 6.1, the estimated consequences of the bundle drop accident are very much lower than those of Configuration 1. However, the consequences are higher than a Reg. Guide 1.25 analysis which would not consider particulates in the gap release source term.

Although the long term storage of spent fuel in the fuel pool is possible, this study considered the transfer of all fuel to an ISFSI. For accident analysis purposes, the Configuration 3 initiator is a tornado generated missile that pierces one cask of the ISFSI. The recommended accident cases assume one assembly is damaged. A high burnup gap release with a small amount of particulates was again assumed. As shown in Table 6.1, the estimated consequences are generally less than the Configuration 2 results.

After all fuel has been removed from the site, the radionuclide inventory that remains, although considerable, cannot be easily dispersed into the environment. Previous studies have estimated very low accident releases that would have negligible offsite and onsite health effects. For the purpose of estimating an onsite accident cost, this study considered the postulated rupture of the Borated Water Storage Tank. The level of released activity, although small, was assumed to require a cleanup. As shown in Table 6.1, BNL estimated a cleanup cost of 110 million dollars for this accident.

**Regulatory Assessment**

After a plant is permanently shutdown, awaiting or undergoing decommissioning, certain regulations, which are based on full power operation, may no longer be applicable. BNL identified a list of candidate regulations (or technical issues) from a screening of 10CFR Parts 0-199. Each of these technical issues was subjected to a detailed review which included federal register notices, SECY memos, NRC policy statements, regulatory guides, standard review plans, NUREG reports, NUREG/CR reports, etc. to develop an understanding of the regulatory bases. The continued applicability of each technical issue was assessed within the context of each spent fuel storage configuration, the results of the consequence analyses, as well as, the expected plant configuration.

The public risk associated with a permanently shutdown nuclear power plant is very different from an operating unit, both in magnitude and content. Accident sequences such as LOCAs and ATWs are no

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*NUREG/CR-4982 used Millstone and Ginna information (Circa 1987) to develop a "snapshot" of plant specific spent fuel pool radionuclide inventories that have since been exceeded.

longer relevant to the defueled facility. Regulations that are designed to protect the public against full power and/or design basis accidents are no longer applicable. Therefore, it is recommended that the following regulations be deleted for all spent fuel storage configurations of the permanently shutdown plant:

- Combustible Gas control (50.44)
- ECCS Acceptance Criteria (50.46)
- Environmental Qualification (50.49)
- Operator Presence at the Controls (50.54 (k))
- Containment Leakage Testing (50.54(0), Appendix J)
- Fracture Prevention Measures (50.60, 50.61, Appendices G and H)
- ATWS Requirements (50.62)
- Loss of All AC Power (50.63)

Other regulations, although based on the full power operating plant, may continue to be partially applicable to the permanently defueled facility. Typically, the scope of these requirements can be reduced to eliminate those that do not pertain to the safe storage of the spent fuel or are no longer necessary to protect the health and safety of the public. The following regulations have been assessed to remain partially applicable for one or more configurations of the permanently shutdown plant:

- Fitness for Duty (Part 26, 55.63(j),(k))
- Technical Specifications (50.36, .36b)
- Fire Protection Program (50.48, Appendix R)
- Quality Assurance Program (50.54(a), Appendix B)
- Operator Staffing Requirements (50.54(m))
- Operator Requalification Program (50.54(i), 55.45, 55.59)
- Security Plan (50.49(p), 70.32, Part 73, Part 73 Appendices B and C)
- Inservice Inspection Requirements (50.55a(g))
- Maintenance Effectiveness* (50.65)

Several technical issues do not fit into these categories. They are discussed below.

We have recommended the continued application of the periodic FSAR update requirement (50.71(e)) to provide a basis for the 50.59 safety evaluations that will be performed when a plant ceases operation. The special nuclear material control requirements of Parts 70 and 74 should continue as long as fuel remains within the plant. The annual fees for the permanently shutdown plant licensees (171.15) should be adjusted to reflect the generic regulatory costs that are directly applicable to their facility type.

The emergency planning and preparedness requirements (50.47, 50.54(q), (t) and Appendix E) and the insurance issues (50.54(w) and Part 140) were evaluated using the accident consequence analyses of this

*Assumes a formal request for permanent cessation of operation after 7/10/96.

study. The estimated consequences for the Configuration 1 accident approximate those of a core damage accident.

It is recommended that all offsite and onsite emergency planning requirements remain in place, with the exception of the Emergency Response Data System requirements of Part 50, Appendix E, VI.

The offsite emergency planning and preparedness (EP) requirements are expected to be eliminated for Configuration 2, based on the results of the generic PWR calculation which estimated a 9 millirem dose at the exclusion area boundary (see Table 6.1).* Part 50 offsite EP requirements can also be eliminated for Configurations 3 (plant only) and 4 because the spent fuel has been transferred to an ISFSI (Part 72 requirements) or transported offsite. Without spent fuel, the plant is not a significant health risk.

It is recommended that the onsite property damage and the offsite liability insurance levels remain at operating reactor levels for the duration of Configuration 1. The consequence analyses of Section 4 support reduced insurance requirements for the remaining configurations.

*However, since plant specific parameters (such as exclusion areas) can vary we recommend that the licensee perform a plant specific evaluation for Configuration 2.

**Table 6.1 Generic PWR Accident Summary[1]**

| Spent Fuel Storage Config. | Accident Timing (yrs after final SD) | Recommended Accident Case | Offsite Consequences | | | | | | Dose at Exclusion Boundary (rem) | Onsite Cleanup Cost ($) |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | Distance (miles) | Prompt Fatalities | Societal Dose (person-rem) | Latent Fatalities | Condemned Land (sq. miles) | Total Cost ($) | | |
| 1 | ~0 | 2L[2] | 0-50<br>0-500 | 0.3<br>0.3 | 4.2E+7<br>7.0E+7 | 16,800<br>28,800 | 156<br>188 | 5.6E+10<br>5.9E+10 | NC | NC |
| 2 | 3.5 (PWR) | Low gap release | 0-50<br>0-500 | 0<br>0 | 3000<br>4000 | 1<br>2 | 0<br>0 | neg. | .009 | 3.2E+7 |
| 3 | 5 | Single best assembly, estimate release | 0-50<br>0-500 | 0<br>0 | 590<br>690 | 0.18<br>0.22 | 0.002<br>0.002 | neg. | 0.472 | 1.2E+7 |
| 4 | 5 | BWST failure | - | - | - | - | - | - | - | 1.1E+8 |

[1]The accident consequences associated with the generic PWR are more severe than the comparable BWR cases.

[2]Rapid zircaloy oxidation involving the last full core offload (and the last normal offload for PWRs) low release fractions assumed.

NC = not calculated; neg = negligible

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# APPENDIX A PREVIOUS ANALYSES OF SPENT FUEL POOL ACCIDENTS

## A.1 DISCUSSION

The Reactor Safety Study[1] considered accidents involving spent fuel. The inventory of material that was potentially at risk was limited to one third of a reactor core. This was consistent with the intention of the routine shipping of spent fuel for reprocessing (or disposal). The Reactor Safety Study concluded that the risk associated with spent fuel storage was extremely small in comparison to that associated with the operating reactor core.

During the Carter administration a federal moratorium halted the reprocessing of spent commercial reactor fuel. Given the absence of away-from-reactor storage facilities or a permanent disposal facility, utilities had no alternative but to store spent fuel at the reactor site. This led to increasingly larger inventories of fuel being stored in reactor spent fuel pools. Modified spent fuel storage racks have also been employed to further increase the ultimate capacities of most reactor spent fuel pools.

A.S. Benjamin and others[2-3] published investigations of the probable course of events following the complete draining of a spent fuel pool. A theoretical model and the computer codes SFUEL and SFUEL1W were developed and employed to analyze the thermal-hydraulic behavior of stored spent fuel assemblies on exposure to air. These studies indicated, that for certain combinations of storage configurations and decay times, freshly discharged fuel assemblies could self heat to a temperature where the air oxidation of the zircaloy fuel cladding would become self sustaining. The additional chemical heat released during clad oxidation, which is comparable to the decay heat, then causes a rapid temperature increase with the resultant failure of the cladding. Additionally, these studies further concluded that for certain conditions, the cladding of freshly discharged assemblies would attain a sufficiently high temperature to heat adjacently located assemblies, with lower decay heat, to the point of "ignition" (self sustaining clad oxidation). The possibility of propagation from assembly to assembly with the involvement of the entire spent fuel pool inventory was not ruled out in all cases.

V.L. Sailor, et al.,[4] reported a study of severe accidents in spent fuel pools. Their investigation provided an assessment of the potential risk from possible accidents in spent fuel pools. The authors describe their effort as a "simplified analysis which followed the logic of a typical probabilistic risk assessment (PRA)." To assess the risk Sailor, et al., quantified the frequencies of initiating events that could compromise the integrity of fuel pool, the probability of system failure conditional on the initiating event, fuel failure occurrence, the magnitudes of radionuclide releases to the environment and the consequences which result from those releases as well as the consequences associated with these releases.

In the Sailor study, two plants were primarily selected for examination on the basis of perceived vulnerability to seismic events. A preliminary screening study using RSS methodology indicated seismic

initiated pool failure was the dominant risk contributor. The selected plants were the Millstone 1 (BWR) and the Ginna (PWR) plants. The operating histories of these plants were used to model, through application of the ORIGIN code, realistic radionuclide inventories present in their respective spent fuel pools, at the time the study was performed.

The accident initiators considered in Sailor's work were loss of pool heat removal capability, structural failure of the pool due to missiles, seismic events or the drop of a heavy load on the pool wall, and the draining of the pool due to pneumatic seal failure. The study concluded accidents which lead to the complete draining of the spent fuel pool caused by loss of cooling, missiles and pneumatic seal failure were very unlikely. However, failures resulting from seismic events and the drop of a heavy load were concluded to be credible, though the frequencies of these accidents was assessed to be quite uncertain. As part of Sailor's study, BNL performed a review of the SFUEL1W models and code. Limited verifications of the code's prediction with the results of small scale experiments performed at SNL were also made. Sailor, et al., concluded that the SFUEL1W code "provides a valuable tool for assessing the likelihood of self-sustaining clad oxidation for a variety of spent fuel configurations assuming the pool has been drained."

Although BNL made at least one modification to the SFUEL1W code, their predictions of critical decay times,* were in good agreement with the earlier published results of the SNL staff.

To estimate the release of radioactivity from the fuel pins, the authors employed the CORSOR code,[5] using the time-temperature histories obtained with the SFUEL1W code. These results are reproduced in Table A.1. The releases are expressed as sets of fractions, which are applied to the total inventory of material involved in the accident. The initial inventory of radionuclides available for release as noted above was calculated with the ORIGIN code using the operating histories of the selected plants. The calculated inventories were a realistic snapshot of the activity present in the spent fuel pools of the selected plants at the time Sailor's study was completed. These inventories are not presented here for several reasons. Both plants investigated were relatively small: 2011 Mw thermal in the case of the BWR and 1520 Mw thermal for the PWR. Continued operation at these plants has also increased their present spent fuel pool inventories. But more importantly, the last one third core discharge was for a normal refueling, and this would represent a significant underestimation of a full core off-load, which was evaluated in the present study.

Offsite accident consequences in NUREG/CR-4982 were calculated with the CRAC2 computer code.[6] Major assumptions used in the evaluation included: a generic site having uniform population density of 100 persons per square mile (approximately the national average); generalized average weather conditions; and the emergency response action being relocation 24 hours after release (criterion 25 rem whole body projected individual dose commitment). The consequences reported, societal dose and

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*The cooling time required to lower the decay heat of freshly discharged fuel assemblies to a point where the self sustaining clad oxidation is unlikely to occur.

interdicted land, are presented in Table A.2. The risk estimates of Sailor's work have been superseded by more recent studies.[7-8] However, it should be noted that to evaluate the risk the authors ultimately selected the consequence results of an accident where the only the last refuelling discharge is involved. In this accident, fire does **not** propagate its way throughout the entire spent fuel pool, but the maximum release fractions were assumed (no credit taken for structures removing activity).

The 1989 report of J. Jo, et al.[7] described a value/impact assessment of various proposed options[2-4] intended to reduce the risk of potential accidents occurring in the commercial nuclear power plant spent fuel pool. As was the case with previous efforts, attention was limited to an operating plant. The risk dominant accidents, source terms and inventory assumptions were identical to those investigated by Sailor, et al. Major differences in the estimation of the offsite consequences existed between these two studies. Jo, et al., used the MELCOR Accident Consequence Code System (MACCS), Version 1.4.[9] This code, developed by Sandia National Laboratory for the NRC, has replaced the CRAC2 code for offsite consequence assessment. The MACCS code has been used exclusively in the preparation of NUREG-1150 and its supporting documentation.[10] Site assumptions which significantly affected the predicted consequences also differed. The Zion site was selected by Jo to represent the "worst" case conditions in regard to population density distributed about a plant site. The actual population distribution, weather conditions, land usage fraction and regional economic data associated with the Zion site were employed. These actual data, coupled with release assumptions of 100 percent pool involvement and the set of maximum fractional releases specified by Sailor, were used to evaluate a **worst** case. For a **best** estimate calculation of accident consequences, the study assumed: only the last refueling discharge is involved in the fire; Zion weather; average land usage and economic data for the state of Illinois; a 95 percent land fraction and a uniform population density of 340 persons per square mile out to 50 miles beyond the plant.* In both cases examined, no planned evacuation was modelled, since this was stated to have only a small effect on total costs and societal doses. However, people were relocated at one day based on projected 7 day dose commitment of 25 rem. (Prior to relocation people were assumed to be engaged in normal activity, which afforded them limited protection from the early dose pathways.) The long term dose limit of 25 rem effective dose equivalent (EDE) employed in this effort was consistent with WASH-1400. The results of these calculations are shown in Table A.3. The public dose and offsite property damage were reported out to 50 miles from the plant. The public doses reported by Jo, et al., are factors of 3.5 and 10 (best estimate and worst case, respectively) higher than those reported by Sailor, et al. The population density assumptions of the latter study ( 340 and 860 persons per square mile versus the 100 used in the Sailor study) account for 98 and 87 percent, respectively, of the observed increases. As such, and notwithstanding consequence codes differences in the release and health effects modeling, the societal dose results of Sailor and the more recent Jo effort appear to be fairly consistent.

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*The average population density for existing plants, circa 1980.[11]

**Table A.1 Estimated Radionuclide Release Fraction During a Spent Fuel Pool Accident Resulting in Complete Destruction of Cladding (Cases 1 and 2)**

| Chemical Family | Element or Isotope | Release Fraction* | |
|---|---|---|---|
| | | Value Used | Uncertainty Range |
| Noble gases | Kr, Xe | 1.00 | 0 |
| Halogens | I-129, I-131 | 1.00 | 0.5-1.0 |
| Alkali Metals | Cs, (Ba-137m) Rb | 1.00 | 0.1-1.0 |
| Chalcogens | Te, (I-132) | 0.02 | 0.002-.02 |
| Alkali Earths | Sr, (Y-90), Ba (in fuel) | $2x10^{-3}$ | $10^{-4}$-$10^{-2}$ |
| | Sr, Y-91 (in clad) | 1.00 | 0.5-1.0 |
| Transition Elements | Co-58 (assembly hardware) | 0.10 | 0.1-1.0 |
| | Co-60 (assembly hardware)** | 0.12 | 0.1-1.0 |
| | Y-91 (assembly hardware) | 0.10 | 0.1-1.0 |
| | Nb-95, Zr-95 (in fuel) | 0.01 | $10^{-3}$-$10^{-1}$ |
| | Nb-95, Zr-95 (in clad) | 1.00 | 0.5-1.0 |
| Miscellaneous | Mo-99 | $1x10^{-6}$ | $10^{-8}$-$10^{-5}$ |
| | Ru-106 | $2x10^{-5}$ | $10^{-6}$-$10^{-4}$ |
| | Sb-125 | 1.00 | 0.5-1.0 |
| Lanthanides | La, Ce, Pr, Nd, Sm, Eu | $1x10^{-6}$ | $10^{-8}$-$10^{-5}$ |
| Transuranics | Np, Pu, Am, Cm | $1x10^{-6}$ | $10^{-8}$-$10^{-5}$ |

*Release fractions of several daughter isotopes are determined by their precursors, e.g., Y-90 by Sr-90, Tc-99m by Mo-99, Rh-106 by Ru-106, I-132 by Te-132, Ba-137m by Cs-137, and La-140 by Ba-140.

**Release fraction adjusted to account for a 100% release of the small amount of Co-60 contained in the zircaloy cladding.

(Reproduced from NUREG/CR-4982)

**Table A.2 CRAC2 Results for Various Releases Corresponding to Postulated Spent Fuel Pool Accidents with Total Loss of Pool Water**

| Case Description | | Whole Body Dose (Man-rem) | Interdiction Area (sq. miles) |
|---|---|---|---|
| 1A. | Total inventory 30 days after discharge 50 mile radial zone | $2.6 \times 10^6$ | 224 |
| 1B. | Total inventory 90 days after discharge 50 mile radial zone | $2.6 \times 10^6$ | 215 |
| 1C.* | Total inventory 30 days after discharge 500 mile radial zone | $7.1 \times 10^7$ | 224 |
| 2A. | Last fuel discharge 90 days after discharge 50 mile radial zone (maximum release frac tion) | $2.3 \times 10^6$ | 44 |
| 2B. | Last fuel discharge 90 days after discharge 50 mile radial zone (minimum release fraction) | $1.1 \times 10^6$ | 4 |
| 2C. | 50% of all fuel rods leak 1 year after discharge 50 mile radial zone | 4.0 | 0.0 |

*Note that the consequence calculations in NUREG-1150 are based on a 50 mile radial zone. Case 1C is given as a sensitivity result.

(Reproduced from NUREG/CR-4982)

**Table A.3 Offsite Consequence Calculations**

| Case | Characterization | Source Term* | Population | Public Health Dose (person-rem) | Offsite Property Damage ($1983) |
|---|---|---|---|---|---|
| 1 | Average case | Last fuel discharge 90 days after discharge | 340 persons/mile$^2$ | $7.97 \times 10^6$ | $3.41 \times 10^9$ |
| 2 | Worst case | Entire pool inventory 30 days after discharge | Zion population (roughly 860 persons/mile$^2$ | $2.56 \times 10^7$ | $2.62 \times 10^{10}$ |

(Reproduced from NUREG/CR-5281)

**Table A.4 Onsite Property Damage Costs Per Accident ($)**

| Item | Best Estimate | Worst Case |
|---|---|---|
| Cleanup and Decontamination | 1.65E8 | 1.65E8 |
| Repair | 7.2E7 | 7.2E7 |
| Replacement power | 8.67E8 | 1.66E9 |
| Total number of operating years remaining | 29.8 years | 29.8 years |
| Number of years plant is out of service | 5 years | 7 years |
| Expected Dollar loss | 8.24E9 | 1.29E10 |

(Reproduced from NUREG/CR-5281)

**Occupational exposure** for a major spent fuel pool accident was assumed in the Jo report to be similar to the estimated occupational exposure, of 4850 man-rem,[11] incurred during the recovery of the Three Mile Island plant. The Jo report stated that "This exposure is small compared to the potential off-site dose impact and more refined quantification appears to be unwarranted."

Onsite property damages were also estimated in the Jo study. The cost of a major spent fuel pool accident was expected to be similar to the cost associated with a Category II severe accident as defined in Reference 13. The estimates provided in the Jo report are reproduced in Table A.4.

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# APPENDIX B DETAILED REGULATORY ASSESSMENT

## B.1 INTRODUCTION

This section provides a detailed assessment of each of the regulations (or technical issues) that may not be fully applicable to permanently shutdown nuclear power plants. This list of candidate regulations was identified from a screening of 10CFR Parts 0-199[1] and is presented in Table B.1. Each of these technical issues was subjected to a detailed review which included federal register notices, SECY memos, NRC policy statements, regulatory guides, standard review plans, NUREG reports, and NUREG/CR reports to develop an understanding of the regulatory bases. The continued applicability of each technical issue was assessed within the context of each spent fuel storage configuration,* the associated safety hazard analysis results, as well as the expected plant status.

With the possible exception of Part 171, "Annual Fees for Licenses," each regulation is ultimately focussed on the protection of public health and safety. However, a particular regulation may not be applicable to a permanently shutdown plant in general, or to a specific spent fuel storage configuration. For example, an exemption from the containment leakage testing requirements of 10CFR50.54(o) for a permanently defueled plant will not impact public health and safety as the plant risk is primarily associated with the spent fuel that is now stored in the spent fuel pool outside the primary containment.

The remainder of this appendix examines each of the candidate regulations of Table B.1. A short discussion of the regulatory background and objective is provided. Our assessment of the continued applicability to each spent fuel storage configuration is stated with additional supporting information, as necessary.

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The spent fuel retrievability requirements for ISFSIs may perturbate the regulatory assessment presented in this appendix. An ISFSI storage method (i.e., NUHOMS or storage only casks) that is presently not licensed for offsite transportation under 10CFR Part 71, may require an operating onsite spent fuel pool to comply with the retrievability requirement of 72.122(l). The BNL recommendations assume: dual purpose cases are used; a NUHOMS transport cask will be licensed; storage only casks (with modifications) can be licensed for transport; or that fuel transfer methods will be licensed that do not require an operating onsite spent fuel pool.

**Table B.1 Assessment of Continued Regulatory Applicability for Permanently Shutdown Nuclear Power Reactors (Summary)**

| Technical Issue | 10CFR Reference[1] | Regulatory Applicability[2-5] Configuration | | | |
|---|---|---|---|---|---|
| | | 1 | 2 | 3[6-9] | 4 |
| Fitness for Duty | Part 26<br>55.53(j), (k)<br>72.194 | P<br>F | N<br>N | N<br>N<br>F | N<br>N |
| Technical Specifications | 50.36, .36b<br>72.26, 72.44 | P | P | P<br>F | P |
| Combustible Gas Control | 50.44 | N | N | N | N |
| ECCS Acceptance Criteria | 50.46 | N | N | N | N |
| Emergency Planning | 50.47, .54(q),(t) App. E<br>72.32 | F | P | P<br>F | P |
| Fire Protection | 50.48, App. R<br>72.122 | P | P | P<br>F | P |
| Environmental Qualification | 50.49 | N | N | N | N |
| QA Program | 50.54(a), App. B<br>Part 72, Subpart G | P | P | P<br>F | P |
| Operator Requalification Program | 50.54(i), 55.45, 55.59<br>72.44(b) Part 72, Subpart I | P | P | N<br>F | N |
| Operator Staffing Requirements | 50.54(k),<br>50.54(m) | N<br>F | N<br>P | N<br>N | N<br>N |
| Containment Leakage Testing | 50.54(o), App. J | N | N | N | N |
| Security Plan | 50.54(p), 70.32, Part 73<br>Part 73 App. B and C<br>72.44(e) Part 72 Subpart H<br>Part 73 | P | P | N<br>F | N |
| Onsite Property Damage Insurance | 50.54(w)<br>Part 72 | F | P | P<br>* | P |

*See discussion in the text.

| Technical Issue | 10CFR Reference[1] | Regulatory Applicability[2-5] Configuration 1 | 2 | 3[6-9] | 4 |
|---|---|---|---|---|---|
| Inservice Inspection Requirements | 50.55a(g) | P | P | N | N |
| Fracture Prevention Measures | 50.60, .61, Apps. G and H | N | N | N | N |
| ATWS Requirements | 50.62 | N | N | N | N |
| Loss of all AC Power | 50.63<br>72.122(k) | N | N | N<br>F | N |
| Maintenance Effectiveness | 50.65 POL before 7/10/96<br>POL after 7/10/96 | N<br>P | N<br>P | N<br>N | N<br>N |
| Periodic FSAR Update Requirement | 50.71(e)<br>72.70 | P | P | P<br>F | P |
| Training and Qualification of Nuclear Power Plant Personnel | 50.120 | P | P | N | N |
| Material Control/Accounting of Special Nuclear Material (including US/IAEA Agreement) | 70.51, .53, 74.13(a), Part 75<br>72.72, .76 | F | F | N<br>F | N |
| Financial Protection Requirements | Part 140<br>Part 72 | F | P | P<br>* | P |
| Annual Fees for Licenses | 171.15<br>171.16 | P | P | P<br>F | P |

**NOTES TO TABLE B.1**

1. 10CFR Parts 0 to 199, revised January 1, 1995.

2. All other regulatory requirements applicable to nuclear power reactors and not listed in this table are assumed to remain in effect, unless addressed by a plant-specific exemption.

3. The spent fuel storage configurations are defined in Sections 2 and 3 of this report. Briefly:
   Configuration 1 - hot fuel in the spent fuel pool
   Configuration 2 - cold fuel in the spent fuel pool
   Configuration 3 - all fuel stored in an ISFSI
   Configuration 4 - all fuel shipped offsite

4. Configuration 1 also assumes the licensee has permanently caused operation and that a confirmatory letter has been issued to prevent refueling the vessel without NRC authorization.

**NOTES TO TABLE B.1 (Cont'd)**

5. F - Regulation continues to be fully applicable for this spent fuel storage configuration.
   P - Regulation is assessed to be partially applicable for this configuration.
   N - Regulation is not considered applicable to this configuration.

6. A permanently shutdown nuclear power plant may store its fuel in an Independent Spent Fuel Storage Installation, before, during, and after the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. This necessitates two entries in Table B.1 for Configuration 3. The first (and second, if applicable) pertains to the plant itself prior to the completion of decommissioning. The last entry examines the Part 72 requirements for the ISFSI.

7. The requirements of Configuration 3 remain applicable until all fuel has been removed from the ISFSI and shipped offsite.

8. In addition to the applicable provisions of Part 72 as noted for Configuration 3, Parts 20, 21, 71, and 73 remain applicable to the transportation of spent fuel from the ISFSI to a HLW repository or MRS.

9. This regulatory assessment assumes an onsite, operating spent fuel pool is not necessary to satisfy the fuel retrievability requirement of 72.122(l). See the introductory section of Appendix B for further information.

## B.2 REGULATORY ASSESSMENT

### Fitness for Duty Program

#### *Background*

The Fitness for Duty Program is contained in Part 26 of Title 10, Code of Federal Regulations. Another reference to the Fitness for Duty can be found in the Operators Licenses Section (10CFR55.53(j),(k)). The licensing requirements for the independent storage of spent nuclear fuel and high level radioactive waste (10CFR72.194) do not require a formal fitness for duty program.

The Fitness for Duty Final Rule was published in the June 7, 1989 Federal Register (54 FR 24468) The Supplementary Information, published with the rule, provided the general background, the need for a rule and a summary of comments on the proposed rule with NRC responses.

The NRC stated that the objective of the rulemaking was to provide reasonable assurance that nuclear power plant personnel were not mentally or physically impaired from any cause which could adversely affect their ability to safely and competently perform their duties. The rulemaking action was taken to significantly increase the assurance of public health and safety. All workers with unescorted assess to the nuclear power reactor protected area, as well as personnel who are physically required to report to the TSC or the EOF under emergency conditions, fall within the scope of this rule.

The associated backfit analysis found that the rule will prove a substantial increase in the overall protection of public health and safety and that the direct and indirect costs of implementation are justified in view of the increased protection. In response to comments on the proposed rule, the NRC reiterated that the Fitness for Duty Rule was limited to nuclear power reactors and they saw no reason to extend the coverage of the rule to other facility types such as non-power test reactors, materials facilities, and special nuclear materials licensees. By extension, one can surmise that the lesser public risk associated with non-power reactors, materials licensees, and independent spent fuel storage installations (ISFSIs) did not warrant the implementation of a fitness for duty program at those facilities.

*Assessment*

Configuration 1, "Hot Fuel in the Spent Fuel Pool" postulated rapid zircaloy oxidation of the spent fuel rods after the loss of pool water inventory. The safety hazard analyses of Section 4 has estimated consequences that are approximately equal to a severe core damage accident. Given the potential magnitude of the consequences, it is appropriate that a formal fitness for duty program, in accordance with the requirements of 10CFR Part 26, remain in place. In recognition of the defueled status of the permanently shutdown plant, and the lack of significant non-fuel sources of public risk,[2,3] It is recommended to reduce the scope of the program to those personnel with unescorted assess to any area that contains equipment necessary to support and maintain continued safe storage or handling of spent fuel. As shown in Table B.1, the Part 26 requirements should remain fully applicable for licensed operators (10CFR55.53(j),(k)).

Configuration 2, "Cold Fuel in the Spent Fuel Pool," has sufficiently low decay heat loads such that the cladding will remain intact even if all spent fuel pool water is lost. Configuration 2 considers the consequences of a dropped fuel assembly. The safety hazard analysis, as discussed in Section 4, shows minimal offsite consequences. On this basis, it appears that the Part 26 requirements for Configuration 2 can be deleted without a significant impact on the public health and safety.

In lieu of long-term storage in the spent fuel pool, a permanently shutdown nuclear power plant may store its spent fuel in an Independent Spent Fuel Storage Installation (ISFSI), before, during, and after the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. Although the postulated accident for Configuration 3 does result in offsite consequences, the results are not dependent on human intervention. Other postulated ISFSI accidents found in the literature[4,5] do not result in significant offsite consequences.

As discussed below, decommissioning accidents, not involving spent fuel, do not have offsite consequences. Therefore, a Part 26 program for Configuration 3 would not significantly impact the health and safety of the public. The requirements of 10CFR72.194 regarding the physical condition of certified ISFSI operation personnel govern.

Configuration 4, "All Fuel Removed from the Site," assumes that all spent fuel has been shipped offsite, including any that might have been stored in an ISFSI. As discussed in Section 4, the postulated accidental radioactive releases to the atmosphere during decommissioning do not pose a significant threat to the onsite workers and/or the public.

Based on the limited consequences associated with Configuration 4, a Part 26 Fitness for Duty program would not have a significant effect on public health and safety.

Although the Fitness for Duty Program requirements may no longer be appropriate for certain spent fuel storage configurations, the record keeping requirements of section 26.71 are still applicable.

## Technical Specifications

### *Background*

Section 50.50 of 10 CFR, "Issuance of Licenses and Construction Permits" provides that each operating license for a nuclear power plant issued by the NRC will contain such conditions and limitations that the Commission deems appropriate and necessary. Operating technical specifications, imposed by Section 50.36 in the interest of the health and safety of the public, are included as Appendix A of the operating license.

Under 10CFR50.36b non radiological environmental technical specifications to protect and monitor the plant's impact on the environment can be included as Appendix B to the license.

Each applicant for an operating license proposes technical specifications for its plant which are then reviewed by the NRC and modified, as necessary,. This process results in a set of plant-specific technical specifications that reflect plant-specific design and siting characteristics. Additional changes, in the form of license amendments, may be granted by the NRC over the operating life of the plant, as appropriate.

### *Assessment*

Very few plants have a defueled mode in their technical specifications. After a permanent cessation of operations issued, the existing technical specifications can be modified to include a permanently defueled mode to reflect the more limited range of postulated accident and radiological consequences associated with a permanently shutdown nuclear power plant. The defueled mode will represent a significant scope reduction in comparison to the operating plant technical specifications requirements. For example,

shutdown margin calculations, (normally required for all tech spec modes) and cooling tower drift or noise monitoring programs would no longer be necessary from a health and safety or an environmental impact perspective.

Since the technical specifications can be very plant specific, it is recommended that the licensee submit an amendment request to reduce the scope of the operating technical specifications and the environmental technical specifications* (or institute a permanently defueled mode) after permanent cessation of operations. Subsequent amendments to the plant technical specifications may be appropriate as the spent fuel decay heat declines (Configuration 2) or if all fuel is moved to an ISFSI** or removed from the site (Configurations 3 and 4, respectively).

## Combustible Gas Control

### *Background*

The combustible gas control requirements are found in 10CFR50.44. These requirements were instituted to "improve hydrogen management in LWR facilities and to provide specific design and other requirements to mitigate the consequences of accidents resulting in a degraded reactor core" [46 FR 58484, 12/2/81].

### *Assessment*

The requirements focus on the capability for: measuring hydrogen concentrations, ensuring a mixed atmosphere and controlling combustible gas mixtures, post LOCA. The concern is that hydrogen generation due to metal water reaction or the radiolytic decomposition of water during a LOCA could result in a detonation or deflagration that could fail primary containment.

Obviously, the post LOCA control of combustible gases inside containment is an operating plant issue. The permanently shutdown plant stores all of its fuel outside containment; the reactor pressure vessel and the primary containment are no longer necessary fission product barriers. Therefore, it is recommended that the requirements of 10CFR50.44 be removed for all four spent fuel configurations for the permanently shutdown nuclear power plants.

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*The technical specifications on effluents for nuclear power reactors (50.36a and Appendix A) continue to remain fully applicable to permanently shutdown plants.

**ISFSIs have their own technical specification requirements under 72.26 and 72.44.

## ECCS Acceptance Criteria

### *Background*

The acceptance criteria for emergency core cooling systems (ECCS) for light water reactors is found in 10CFR50.46. This section requires that the ECCS be designed to limit post LOCA peak cladding temperature, clad oxidation and hydrogen generation to specified values and provide for long-term cooling. Acceptable ECCS evaluation models must address the sources of heat during a postulated LOCA, clad swelling or rupture, blowdown phenomena, etc. Although this section is primarily addressed during the design phase, operating license holders are required to estimate the effect of a change or an error in the ECCS evaluation model or the model application. Section 50.46(a)(3) specifies the reporting and reanalysis requirements, which are dependent on the magnitude of the error or change.

### *Assessment*

The purpose of these requirements is to ensure that the ECCS design can, and continues to be able to, mitigate the design basis LOCA throughout the operating life of the plant. Without fuel in the vessel, a permanently shutdown plant could make changes to its ECCS systems without a significant public health and safety impact, yet an ECCS re-evaluation could be required. Therefore, the ECCS acceptance requirements of 10CFR50.44 may be deleted for all spent fuel storage configurations of the permanently shutdown plant.

## Emergency Planning

### *Background*

The emergency preparedness requirements for nuclear power reactors are contained under 10CFR50.54, "Conditions of Licenses." Paragraph (q) requires that a licensee, authorized to possess and operate a nuclear power reactor, follow and maintain in effect emergency plans which meet the standards of Section 50.47(b) and Appendix E to Part 50. Paragraph (t) of 50.54 emphasizes the revision and maintenance of the emergency preparedness program and requires an annual independent review. Section 50.47(b) presents sixteen requirements for offsite and onsite emergency response. Appendix E to Part 50 generally augments the requirements of 50.47(b).

Due to the lower inherent risk to the public, other facilities licensed by the NRC typically have less stringent emergency preparedness (EP) requirements than nuclear power reactors. For example, research reactors and special nuclear materials licensees are also subject to the requirements of Appendix E to Part 50. However, the size of the emergency planning zone for these facilities and the degree of compliance to the requirements of Appendix E are determined on a case by case basis. Materials license applicants, under 10CFR30.32(i) with quantities of radioactive material in excess of Appendix C to Part 30 must furnish either:

- An evaluation showing that the maximum dose to a person offsite due to a radioactive release would not exceed one rem effective dose equivalent or five rems to the thyroid.

- An emergency plan for responding to the release of radioactivity.

*Assessment*

The estimated offsite consequences of a rapid zircaloy oxidation event in the spent fuel pool dictate the continuance of all nuclear power reactor emergency preparedness regulatory requirements* for Configuration 1, "Hot Fuel in the Spent Fuel Pool."

Section 4 of this report developed consequence estimates based on generic BWR and PWR plant parameters, source term assumptions and recommended accident cases. The recommended accident case for Configuration 2 had an estimated dose at the exclusion area boundary (0.4 miles) of 9 millirem for the generic PWR. This dose is well below the EPA Protective Action Guide (PAG) whole body dose of 1 rem at the exclusion area boundary. Since this dose estimate is based on generic plant assumptions (such as the exclusion area boundary, it is recommended that the permanently shutdown plant perform a plant specific evaluation for Configuration 2 and specify sufficiently sized emergency planning zone (EPZ) so that the EPA PAGs are not exceeded at the EPZ boundary. Based on our generic calculations for Configuration 2 Section 4.2, BNL believes a permanently shutdown plant EPZ can be reduced so that it resides entirely within the former full power exclusion zone, i.e., within the site boundary.

Section 4 has also stated that decommissioning accidents that do not involve spent fuel do not pose a significant health risk to the public. Therefore, offsite emergency planning is not required for Configurations 3 (plant only) and 4.

It is recommended that the permanently shutdown licensee apply for exemptions from the following *offsite* emergency planning requirements for Configurations 2,3, (plant only) and 4:

- The early public notification requirements of 50.47(b)(5) and Appendix E.IV.D.3.
- The periodic dissemination of emergency planning information to the public (50.47(b)(7) and Appendix E.IV.E.8).
- Offsite emergency facilities and equipment such as the EOF, and the emergency news center (50.47(b)(8), Appendix E.IV.E.8).
- Offsite radiological assessment and monitoring capability, including field teams (50.47(b)(9)).
- Periodic offsite drills and exercises (50.47(b)(14), Appendix E.IV.F.3).
- Licensee headquarters support personnel training (50.47(b)(15), Appendix E.IV.F.b.h).

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*except the Emergency Response Data System Requirements of Part 50, Appendix E, VI.

The NRC has recently issued a final rule [60 FR 32430, 6/22/95]. The emergency planning requirements for a typical, storage only ISFSI are provided in paragraphs 72.32 (a), (c) and (d).

Onsite emergency planning requirements should remain applicable for all spent fuel storage configurations.

## Fire Protection

### *Background*

Section 50.48 of 10CFR states, "each operating nuclear power plant must have a fire protection plan that satisfies Criteria 3 of Appendix A of this part." Criterion 3 states that fire detection and fighting systems of appropriate capacity and capability are required to minimize the effects of fires on structures, systems, and components important to safety. Section 50.48 further states that basic fire protection guidance provided in two documents: Branch Technical Position APCSB 9.5-1 and its Appendix A. The appropriate document is dependent on the plant's status as of July 1, 1976. The Branch Technical Position (BTP) APCSB 9.5-1 is applicable to new plants docketed after that date, while Appendix A to the BTP addresses older plants that were operating or under design or construction prior to 7/1/76.

### *Assessment*

Although the emphasis of both these documents is the preservation of the safe shutdown capability during and after a fire, the guidance recognizes other sources of risk that are not related to reactor shutdown or in vessel decay heat removal. Appendix A to BTP APCSB 9.5-1 requires:

- The fire protection program for new fuel storage areas (and adjacent fire zones that could affect the fuel storage zone) be fully operational before fuel is received at the site.
- Fire protection and automatic detection for the spent fuel pool area.
- Radwaste building detection and protection.
- Materials that contain radioactivity must be stored in closed metal tanks or containers, away from ignition sources of combustibles.

Each licensee has a fire protection program that, in addition to safe shutdown requirements, has fire brigade training requirements, administrative procedures and controls, and detection and suppression requirements for plant areas that contain radioactive inventories with potential offsite consequences. For Configurations 1, 2, 3, (plant only) and 4, we recommend eliminating those requirements directly related to safe shutdown capability. Additional reductions in the scope of the 50.48 fire protection program can be examined on a plant-specific basis.

ISFSIs, under spent fuel storage Configuration 3, are subject to the fire protection requirements of Section 72.122.

## Environmental Qualification

### *Background*

The Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants (10CFR50.49) was published as a final rule in the January 21, 1983 Federal Register (48FR2729). The supplementary information provided with the rule states:

> *The scope of the final rule covers that portion of equipment important to safety commonly referred to as "safety related".... Safety-related structures, systems, and components are those that are relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10CFR Part 100. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences; design basis accidents; external events; and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) above.*

### *Assessment*

The EQ rule is clearly limited to electrical equipment that must function during design basis events. In response to comments on the final rule, the Commission stated that the EQ rule does not cover the electrical equipment located in a mild environment. With the permanent cessation of operations, the design basis accidents of the FSAR are limited to Section 15.7, Radioactive Release from a Subsystem or Component. The harsh environment associated with loss of coolant accidents is no longer applicable. Therefore, 10CFR50.49 can be deleted for the permanently shutdown plant.

## Quality Assurance (QA) Program

### *Background*

The plant-specific QA program that implements the Part 50 Appendix B QA requirements is described or referenced in the Safety Analysis Report per 10CFR50.34(b)(6)(ii). Under paragraph (a) of "Condition of Licenses (50.54)," the licensee is required to implement the QA program described (or referenced) in the SAR. Furthermore, paragraph (a)(3) requires NRC submittal and approval of any proposed changes that reduce the commitments in the previously accepted QA program.

*Assessment*

The permanently defueled plant can make selected changes to its operating based QA program without impacting public health and safety. As previously discussed in the technical specification section, each plant should evaluate the scope of their QA program and submit the revisions that are appropriate to their facility and mode of spent fuel storage for NRC approval. Perhaps R.G. 1.33 can be revised (or another RG issued) to address the QA program for the PSD plants.

## Operator Requalification Program

*Background*

Section 54(i) of 10CFR Part 50 requires an operator requalification program that meets the requirements of 10CFR55.59(c). The licensee may not decrease the scope of the program, except as authorized by the Commission.

*Assessment*

Part 55 states the requirements for granting and maintaining operator's licenses and is oriented toward operating nuclear power reactors. As a consequence, portions of this section are not applicable to a permanently defueled facility. The following sections should be revised to eliminate those regulatory requirements that solely pertain to operating nuclear power reactors:

55.41, 55.43, 55.45(a), 55.59(c) - Written examinations, operating tests, and requalification program requirements should reflect the permanently defueled plant configuration and the accidents that are applicable to the permanently shutdown facility.

55.45(b) - The operating tests for a permanently defueled plant should be administered in a plant walk-through. Simulation facilities are designed for operating power reactors, have limited usefulness for the defueled configuration, and should not be required for the administration of operating tests. In addition, Section 55.53(k) should be revised to reflect any modifications to the fitness for duty program that may be adopted for the permanently shutdown nuclear power reactor.

When all fuel is removed from the plant, either to an ISFSI (Configuration 3) or offsite (Configuration 4) there is no longer any need for operators licensed under Part 55, and the requalification program can be terminated.*

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*As discussed in Section D.1, this regulatory assessment assumes an operating onsite spent fuel pool is not necessary for fuel retrievability. Therefore, licensed fuel handlers are not necessary for Configuration 3.

## Operator Staffing

### *Background*

The licensed operator staffing requirements for nuclear power reactors are delineated in Sections 50.54(k) and (m).

Paragraph (k) requires a licensed operator to be present at the controls at all times during the operation of the facility. A nuclear power unit is considered to be operating when it is in a mode other than cold shutdown or refueling. By extension, the permanently defueled condition does not require a licensed operator to be continuously present at the controls.

Paragraphs (m)(2)(i) presents onsite licensed operator staffing requirements for nuclear power reactors. The requirements are based on the number of units operating (i.e., not in cold shutdown or refueling) at a site and the number of control rooms. However, onsite staffing is required for non-operating units.

### *Assessment*

The onsite staffing requirements of Section 50.54(m) (2)(i) should remain in effect for Configuration 1. Our concern is the continued ability to: recover from off-normal events (such as the loss of fuel pool cooling) and activate the unit(s) emergency plan. The lower decay heat of the fuel assemblies in Configuration 2 subject to the same concern as Configuration 1. There is a long time for recovery from most off normal events.* Therefore, it is not necessary to require continuous operator staffing onsite unless spent fuel or other objects are being moved within or above the spent fuel pool, or other work is in process that poses a potential near term challenge to fuel cladding integrity. Since Configurations 3 and 4 do not require licensed operators, other personnel would have to be charged with the emergency plan responsibilities.

## Containment Leakage Testing

### *Background*

Conditions of Licenses, 10CFR50.54, Paragraph (o) states that primary reactor containments for water cooled power reactors are subject to the requirements of Part 50, Appendix J. This appendix requires periodic testing to verify the leaktight integrity of the primary containment and those systems and components which penetrate the containment.

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*The representative accident sequence, a fuel assembly drop assumes an operator is present.

*Assessment*

The primary containment of an operating plant is one of several fission product barriers designed to protect the public's health and safety in the event of an accident. In contrast to an operating plant, a permanently defueled facility stores all of its fuel outside containment. The defueled containment is not a source of public risk; previous decommissioning studies[2-3] have determined that there are not significant offsite consequences associated with accidents that do not involve spent fuel. Therefore, the continued maintenance of containment leakage integrity does not enhance public health and safety and it is recommended that these testing requirements be eliminated for the permanently shutdown plant.

## Security Plan

*Background*

As part of the "content of applications" of Section 50.34, applicants for a Part 50 license are required to submit a physical security plan and a safeguard contingency plan. The physical security plan addresses vital equipment, vital areas, and isolation zones and also demonstrates the applicant's compliance with the requirements of Part 73.

The safeguards contingency plan includes plans for dealing with threats, thefts, and radiological sabotage of special nuclear material in accordance with the criteria of Part 73, Appendix C, Section 50.54(p) "Conditions of Licenses", requires prior Commission approval of any changes that would decrease the effectiveness of the security plan,* the guard training and qualification plan, and the submitted portion of the safeguards contingency plan Part 73 and the associated Appendices B and C provide physical protection requirements, access authorization requirements, general criteria for security personnel and safeguards contingency plan criteria for Part 50 licensees.

Independent Spent Fuel Storage Installations also have similar requirements for the ISFSI physical security, guard training and safeguards contingency plans under Section 72.44(e), Part 72 Subpart H, Part 73, and Part 73 Appendix C.

*Assessment*

The intent of the physical security, guard qualification and training, and the safeguards contingency plan is to protect the facility against radiological sabotage and to prevent the theft of special nuclear material. In comparison to operating units, permanently shutdown plants have a limited number of vital areas that are necessary for the protection of those systems required to support spent fuel cooling and storage.

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*Changes that do not decrease the safeguards effectiveness of the aforementioned plans may be made without prior Commission approval.

For permanently shutdown nuclear power plants with fuel storage in the spent fuel pool (Configurations 1 and 2), the use of license amendment requests is recommended to reduce the scope of the security plan with regard to the number and extent of vital areas and equipment.* When the fuel is moved to an ISFSI or offsite (Configurations 3 and 4, respectively) there is no longer any need for the physical security, safeguards contingency or guard qualification and training plans for the permanently shutdown facility.** Please note that the ISFSI has physical security requirements under Part 72 Section 72.44(e), and Subpart H which are independent of the plant status. Under Configuration 4, all spent fuel will be shipped offsite and will become the responsibility of the DOE.

## Onsite Property Damage Insurance

### *Background*

The onsite property damage requirements for nuclear power plants are found in 10CFR 50.54(w). Each licensee is required to have a minimum coverage limit of $1.06 billion or whatever amount is generally available from private sources, whichever is less. This insurance must be dedicated to the expenses associated with returning and maintaining the reactor in a safe and stable condition in the event of an accident and, removing or controlling onsite radioactive contamination such that personnel exposure limits are consistent with the occupational exposure limits of 10CFR Part 20. In the event of an accident with estimated cleanup costs above a threshold of $100 million, paragraph 50.54(w)(4) provides for an automatic prioritization of stabilization activities.

The onsite property damage insurance requirement was instituted in March, 1982 (47FR 13750) and became effective on June 29, 1982. This regulation has been amended several times over the years. During the amendment processes, the Commission provided its views in several areas that are germane to the permanently shutdown plant. These are:

- the purpose of the regulation,
- the required amount of insurance and the updating mechanism, and
- the $100 million threshold for automatically determining stabilization priorities.

Each of these areas is discussed below. The regulatory intent is illustrated with cites from the appropriate Federal Register Notices. The Commission's philosophy is then summarized and applied to the PSD plant.

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*This reduction in the scope of the program could also conceivably reduce the size of the security force and procedures.

**References 2 and 3 and the consequence analysis for Configuration 4 (Section 4.4 of this report) indicate that once all fuel is removed the predicted offsite releases of accidents that could occur during the decommissioning process are much less than the 10CFR Part 100 limits.

## The Purpose of the Regulation

The onsite property damage insurance requirement of 10CFR 50.54(w) was adopted as a final rule in 1982 (47FR 13750, March 31, 1982). As part of this Federal Register Notice, the public comments on the proposed rule were discussed. Several commenters suggested that the rule apply only to insurance covering decontamination of a facility suffering an accident and not to "all risk" property damage insurance. The Commission agreed, stating:

> *"Because decontamination insurance is the Commission's only concern from the point of view of protecting public health and safety, coverage to replace the existing facility on an "all risk" basis is beyond the scope of the Commission's authority."*

This position has been reaffirmed in two subsequent amendments to the regulation (52FR 28963 8/5/87, 55FR 12163 4/20/90. The 1987 amendment also introduced a decontamination priority which established a priority for stabilizing the reactor after an accident to prevent any significant risk to the public health and safety.

## The Required Amount of Property Damage Insurance and the Updating Mechanism

When the onsite property insurance requirement, 10CFR 50.54(w), was originally instituted (47FR 13750, 3/31/82), the Commission required licensees to "take reasonable steps to obtain onsite property damage insurance available at reasonable costs and on reasonable terms from private sources".* The minimum coverage limit was specified as both:

1. the maximum amount of property insurance offered as primary coverage by either American Nuclear Insurers/Mutual Atomic Energy Reinsurance Pool (ANI/MAERP) or Nuclear Mutual Limited (NML) - $500 million, and

2. any excess coverage in amount no less than that offered by either ANI/MAERP - $85 million or Nuclear Electric Insurance Limited (NEIL) - $435 million.

Thus, the minimum required was originally $500 million primary coverage and $85 million excess coverage. By buying both excess layers, many licensees purchased a total of $1.02 billion in onsite property damage insurance (49FR 44646, 11/8/84). The Commission did not quantify a required insurance value at that time. The minimum requirement was viewed as a reasonable amount of insurance, pending the completion of a study evaluating the cleanup costs of accidents of varying severity. That study was issued as NUREG/CR-2601, "Technology Safety and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents".[9]

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*Or to demonstrate an equivalent amount of protection

NUREG/CR-2601 evaluated cleanup costs following three full power accidents of varying severity at two reference light water reactors. The scenario 1 accident is postulated to result in 10% fuel cladding failure, no fuel melting, moderate contamination of the containment structure, but no significant physical damage to buildings and equipment. The scenario 2 accident is postulated to result in 50% fuel cladding failure, a small amount of fuel melting, extensive radioactive contamination of supporting buildings, and minor physical damage to buildings and equipment. The scenario 3 accident is postulated to result in 100% fuel cladding failure, significant fuel melting and core damage, severe radioactive contamination of the containment structure, moderate radioactive contamination of supporting buildings, and major physical damage to structures and equipment. A TMI-2 type accident was assumed in the study to be of intermediate severity (scenario 2).

The cleanup costs established in the report ranged from $105.2 million to $404.5 million for the reference PWR and from $128.5 million to $420.9 million for reference BWR. Although these costs are considerably lower than the roughly $1 billion estimated to be required to cleanup TMI-2, the NRC noted (52FR28963 8/5/87) that the estimates do not include several TMI cost components such as, inflation during the cleanup, additional decontamination of the containment building, and the cost of facility stabilization. These additional cost considerations cause the NUREG/CR-2601 cost estimates to increase to $1.06 billion for the most severe accidents studied and somewhat less for a TMI-2 type accident.

One conclusion the NRC drew from this study was that the minimum insurance requirement of $585 million would be insufficient for some accidents. Accordingly, the NRC amended 10CFR 50.54(w) (52FR 28963, 8/5/87) to require power reactor licensees to maintain at least $1.06 billion of onsite property damage insurance. The NRC noted that previous exemptions from the full amount required by 10CFR 50.54(w) were still valid. These exemptions were granted to four licensees of small reactors based on plant specific analyses of accident costs. The NRC stated:

> *"Increasing the required amount of insurance based on general technical studies in no way negates the continued validity of the specific studies upon which the existing exemptions were based."*

The August 5, 1987 Federal Register Notice also presents a summary of comments on the method of future adjustment of the insurance requirement. The NRC agreed with many commenters that an adjustment formula tied to a measure of inflation (e.g., the Consumer Price Index or the Handy-Whitman Construction Index) would not accurately reflect decontamination cost changes. Although it is expected that nuclear power reactor licensees will purchase the maximum amount of insurance that is reasonably available, the NRC reserves the right to perform periodic analyses to determine changes in accident recovery costs and to conduct rulemaking based on these analyses.

## The Threshold for Automatically Determining Stabilization Priorities

In response to the 1987 final rule on changes in property insurance requirements, several petitions for rulemaking (noticed in 53FR 36335, 9/19/80) were received that requested clarification of the decontamination and stabilization priorities. As part of that rulemaking (55FR 12163, 4/2/90), the NRC amended 50.54(w)(4) to require dedication of insurance proceeds to decontamination and stabilization activities only if the estimated costs exceeded $100 million. This cutoff was viewed as a relatively minor accident where the availability of funds for stabilization decontamination activities is not considered to be an issue.

However, the Commission stated in this rulemaking that if disputes over the stabilization and decontamination process arise, the Rules of Practice under 10CFR Part 2 provide adequate procedures to resolve any issues.

### Summary

This background discussion establishes that the purpose of 10CFR 50.54(w) is to protect health and safety in the unlikely event of an accident at a nuclear power plant. The minimum insurance requirement to assure post-accident recovery is based on the estimated stabilization and decontamination costs developed in NUREG/CR-2601 for two reference plants. Since it is not the Commission's intent to require more insurance coverage than is necessary for these purposes, licensees of smaller reactors have been granted exemptions from the full insurance requirement based on plant specific analyses that demonstrate lower cleanup costs. Finally, the NRC retains the authority to establish accident recovery and cleanup priorities, regardless of the estimated stabilization and decontamination costs.

Clearly the development of lower onsite property damage insurance requirements for the PSD plant is consistent with the intent of the regulation.

### *Assessment*

Section 4 of this report developed accident consequence estimates for the four spent fuel storage configurations that were assessed for this program.

Configuration 1, "Hot Fuel in the Spent Fuel Pool," postulated rapid zircaloy oxidation of the spent fuel rods after the loss of the pool water inventory. The safety hazard analysis (Section 4) has estimated consequences that are approximately equal to a severe core damage accident. Given the potential magnitude of the consequences, it is appropriate that the onsite property damage insurance requirements of 10CFR 50.54(w) remain fully applicable for Configuration 1.

Configuration 2, "Cold Fuel in the Spent Fuel Pool," has sufficiently low decay heat loads such that the cladding will remain intact even if all spent fuel pool water is lost. Configuration 2 considers the

consequences of a dropped assembly. The Configuration 2 onsite cleanup costs has been estimated at $24 million.

In lieu of long term storage in the spent fuel pool, a permanently shutdown nuclear power plant may store its spent fuel in an Independent Spent Fuel Storage Installation (ISFSI), before, during, and after, the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. The postulated accident for Configuration 3 is a non-mechanistic breach of the ISFSI which damages a singleBWR or PWR fuel assembly.* The Configuration 3 onsite cleanup cost is estimated at $12 million.

Configuration 4, "All Fuel Removed from the Site," assumes that all spent fuel has been shipped offsite, including any that might have been stored in an ISFSI. As discussed in Section 4, the postulated accidental radioactive releases to the atmosphere during decommissioning do not pose a significant threat to the onsite workers or the public. For the purpose of estimating onsite accident cleanup costs, the postulated scenario for Configuration 4 is the rupture of the borated water storage tank. Approximately 450,000 gallons of slightly radioactive water is released causing soil contamination. The estimated cleanup cost is $110 million.

## Inservice Inspection and Testing ISI and IST Requirements

### *Background*

10CFR50.55a, Codes and Standards, require that ASME Code Class 1, 2, and 3 pumps, valves, vessels, piping, and supports meet the testing and examination requirements set forth in Section XI of the ASME Boiler and Pressure Vessel Code. Each licensee is required to update and submit their ISI and IST Programs every ten years to the edition and addenda referenced in 10CFR50.55a(b), 12 months prior to the start of the 10 year interval. The initial interval begins at the issuance of the operating license. Section XI provides testing requirements to verify the operational readiness of pumps and valves and the structural integrity of pressure retaining components and their supports.

The ISI and IST Programs contain a plant-specific list of the applicable components, code classification, code category, examinations or tests to be performed, and the frequency and schedule of examination or testing. When the code requirements are impractical, for instance due to plant design, or would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety, the regulations permit alternatives to be used when authorized by the Commission.

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*This consequence estimate may not envelope sabotage scenarios which could conceivably involve a greater radionuclide release. These scenarios are safeguard information. The information on radionuclide release (if any) is not available to BNL.

*Assessment*

Each licensee is required to determine the ASME Code Class 1, 2, and 3 components and prepare and ISI and IST program for these components. Each program is plant specific depending on the design of the plant and the classification of components. The classification may be determined based on Regulatory Guide 1.26, NUREG-0800, or the ANSI/ANS Standards N52.1 and 51.1, depending on the age of the lant and the agreements made with the NRC. The systems important to the permanently defueled plant are radiation monitoring, fuel building, HVAC, and spent fuel pool cooling cleanup. The ASME Boiler and Pressure Vessel Codes do not address instruments and controls such as radiation monitoring. Fuel building HVAC, and spent fuel pool cooling systems may be included in the IST programs, depending on whether they perform a design basis safety-related function. Non-safety related components are not required to be examined or tested in accordance with the Code. Additionally, some plants may not include HVAC systems in the ISI/IST programs because they do not contain water, steam, or radioactive waste.

It is recommended that licensees of permanently shutdown plants reduce the scope of the ISI and IST programs to eliminate those systems that do not support spent fuel storage and handling (including cooling and cleanup) and HVAC. Although the revised program should be submitted to the NRC, approval is not necessary, unless relief requests are revised or added.

## Fracture Prevention Measures

*Background*

Sections 50.60, 50.61, and Appendices G and H to Part 50 specify fracture toughness requirements and material surveillance programs for the reactor coolant pressure boundary of light water reactors. The intent of these regulations is to maintain reactor coolant pressure boundary integrity by assuring adequate margins of safety during any condition of normal operation (including anticipated operational occurrences).

*Assessment*

Once the permanently shutdown plant has been completely defueled, the measures required by these regulations are no longer necessary. These requirements can be eliminated for all spent fuel storage configurations without impacting the health and safety of the public.

## ATWS Requirements

### *Background*

The purpose of 10CFR50.62 is to require improvements in the design and operation of light water cooled nuclear power plants to reduce the likelihood of RPS failure following anticipated operational occurrences. This regulation also requires improvements in the capability to mitigate the consequences of an ATWS event.

### *Assessment*

Although ATWS can be a significant contributor to operating plant risk, it is not applicable to permanently shutdown plants where fuel is stored in subcritical arrays. This regulation can be eliminated for all spent fuel storage configurations of the permanently defueled plants without impacting public health and safety.

## Loss of All AC Power Requirements

### *Background*

The loss of all AC power requirements Station Blackout Rule is found in 10CFR50.63. The regulation requires that all light water cooled nuclear power plants be capable of withstanding a complete loss of AC power for a specified duration and maintain reactor core cooling during that period. The NRC intent is to provide further assurance that a loss of both the offsite and onsite emergency AC power systems will not adversely affect public health and safety.

The Station Blackout (SBO) rule was published in the June 21, 1988 issue of the Federal Register (53FR23203). The supplementary information provided with the rule indicates that the purpose of this regulation is to explicitly require that nuclear power plants be designed to insure that core cooling can be maintained for a specific duration (coping period) without onsite or offsite AC power. The coping period can range from two to sixteen hours depending on the plant-specific design and the site characteristics.

### *Assessment*

The objective of the rule is to reduce the risk of severe accidents resulting from SBO by maintaining highly reliable AC electric power systems and, as an additional defense in depth, assuring that plants can cope with a loss of all AC power for some period of time. The goal is to maintain the core damage frequency contribution of SBO to about $10^{-5}$/reactor year.

Although the rule is oriented toward core damage, the objective of reducing severe accident risk due to SBO can be applied to a permanently defueled plant.

Based on the analysis in NUREG/CR-1353,[6] a total loss of spent fuel cooling would allow over 40 hours of boiloff before any spent fuel would be exposed. This time is well in excess of the maximum coping period required by the rule. The long period before fuel damage occurs allows ample time for offsite power recovery or fuel pool makeup.* BNL has estimated a fuel damage frequency of 5E-7 (with credit for one emergency diesel generator (EDG)) and 4E-5 (no EDGs credited) for an extended loss of all AC power.

BNL believes that permanently shutdown nuclear power plants meet the intent of 10CFR50.63. For consistency with Reg. Guide 1.155, we recommend that the existing (operating based) SBO plant procedures and training be revised to reflect the storage of all fuel in the spent fuel pool (Configurations 1 and 2).

The ISFSI of Configuration 3 should fully conform to the requirements of Section 72.122(k), however since all fuel has been removed from the plant (Configurations 3 and 4) the requirements of 10CFR50.63 are not applicable.

## Maintenance Effectiveness

### *Background*

The NRC amended its regulations under 10CFR50.65 to require commercial nuclear power plant licensees to monitor the effectiveness of maintenance activities on safety significant plant equipment. The intent is to minimize the likelihood of failures and events caused by the lack of effective maintenance. The rule will require that licenses:

- Perform annual evaluation of the effectiveness of the maintenance program.
- Assess the overall impact of monitoring and maintenance activities (which require taking equipment out of service) on the performance of safety functions.

The rule will become effective on July 10, 1996.

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*Reference 6 has estimated a 24 hour recovery period for actions that require access to the spent fuel pool. These could include the use of the fire protection system to provide pool makeup. Remote recovery actions, such as offsite power recovery, are not limited by the auxiliary building radiation levels and must be accomplished before boiloff exposes the fuel.

*Assessment*

Section 50.65, paragraph b (scope of the monitoring program) includes safety-related structures, systems, and components that are relied upon to remain functional during and after design basis events to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to 10CFR Part 100 guidelines. Also included within the scope of the maintenance effectiveness program are non-safety related structures, systems, and components (SSCs) that are relied upon to mitigate accidents. Furthermore, draft regulatory guide DG-1001 [DG-1001, 8/1/89] clarifies the scope of the rule as including "SSCs in the balance-of-plant that would significantly impact safety or security."

Using the draft regulatory guide and other industry guidance each licensee will develop a prescriptive maintenance effectiveness program to meet the intent of the rule.

Plants that have formally ceased operations prior to July 10, 1996 (the effective date of the rule) are not expected to have implemented a maintenance effectiveness program. It is recommended that these facilities be exempted from the requirements of the rule.

Plants that operate after July 10, 1996 should have a maintenance effectiveness program in place. The scope of the program will vary from plant-to-plant based on plant-specific design and operating attributes. When a plant is permanently shutdown many of these structures, systems, and components can be removed from the maintenance effectiveness program. For these plants, the scope of the maintenance effectiveness program can be reduced to reflect the permanently shutdown plant configuration, i.e., it would only apply to the structures, systems, and components necessary to support safe fuel storage in the spent fuel pool (Configurations 1&2).

The requirements of Section 50.65 are not applicable to spent fuel storage Configurations 3 and 4.

## Periodic FSAR Update Requirements

*Background*

10CFR50.71(e) requires NPP licensees to file FSAR revisions annually or six months after each refueling outage (provided the interval between successive updates to the FSAR does not exceed 24 months). The updated FSAR shall "include the effect of all changes made in the facility or procedures described in the FSAR all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question all analyses of new safety issues performed by or on behalf of the licensee at Commission request."

The NRC position on the continued applicability of 50.71(e) to permanently shutdown plants appears to be evolving. Schedular exemptions from 50.71(e) have been issued to PSD licensees in the past.[7]

However, more recently, the Yankee Nuclear Power Station received an exemption from the FSAR update requirements.[8]

*Assessment*

After a decision to permanently shutdown a facility has been formalized with the NRC, a licensee may begin making extensive changes to plant structures, systems, and components that are no longer necessary. Each of these changes will require a 50.59 safety evaluation which in turn requires a FSAR review. The continuance of the FSAR update requirement will provide a somewhat current plant reference source for future safety evaluations and will also continue to serve as a licensing document. In the supplemental information provided as part of the Final Rule [45FR30614, May 9, 1980] the scope of the rule was specifically extended to include older plants without FSARs including the Indian Point 1 and Humboldt Bay plants that were permanently shutdown at the time. In addition, wenote the periodic* FSAR update requirements for ISFSIs, a passive storage system, without the support systems required for fuel storage in the spent fuel pool. It is recommended that the FSAR update requirements of 50.71(e) be maintained for all spent fuel storage configurations, with schedular exemptions as necessary to encourage a timely submittal that documents the plant at major decommissioning milestones. However, the scope of the document is expected to be reduced to reflect the decommissioning process, i.e., the removal of plant systems, structures, components, and procedures, that are no longer necessary from a health and safety perspective. The ISFSI update requirements of 70.72 remain, although for consistency, a biennial update period should be considered.

## Training and Qualification of Nuclear Power Plant Personnel

*Background*

In 1993 the NRC amended its regulations [58 FR 21904, 4/26/93] to require that each applicant and each holder of a license to operate a nuclear power plant establish, implement, and maintain a training program. The new requirement, 10CFR 50.120, uses a systems approach to training to ensure nuclear power plant personnel will be qualified to operate and maintain the facility in a safe manner for all modes of operation.

The rule requires training and qualification of the following nuclear power plant personnel:

- Non-licensed operator
- Shift supervisor
- Shift technical advisor

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*10CFR72.70 currently requires an annual FSAR update for ISFSI licensees. The similar requirement for Part 50 licensees was revised from an annual to a refueling outage basis not to exceed 24 months. (57FR39353, 8/31/92).

- Instrument and control technician
- Electrical maintenance personnel
- Mechanical maintenance personnel
- Radiological protection technician
- Chemistry technician
- Engineering support personnel

Licensed operators, such as control room operators and senior control room operators, are not covered by this rule and will continue to be covered by 10CFR Part 55. Because some senior control room operators may also be shift supervisors, only those aspects of training related to their shift supervisor function are covered by this rule.

As part of the public comments to the proposed Rule, several commenters recommended that facilities undergoing decommissioning, where all fuel has been permanently removed from the reactor vessel, or those with a possession only licensee, not be subject to this Rule. The Commission disagreed, stating that the provisions of the Rule are applicable to all Part 50 licensees. The Commission maintained that the systems approach to training embodied in the Rule will ensure that training programs are revised to reflect changing plant conditions. Permanent changes to the plant (i.e., decommissioning) that make some or all of the existing training programs unnecessary can be addressed by the exemption process. Since the public risk associated with the permanently shutdown nuclear power plant is associated with the spent fuel, it is recommended that the requirements of 50.120 continue for Configurations 1 and 2 for only those personnel that are responsible for fuel handling and the continued safe storage of the spent fuel.

As shown in the safety hazard analyses of Section 4, after the spent fuel has been moved to an ISFSI or offsite, the risk to the public is negligible. The training and qualification requirements of 50.120 can therefore be removed for Configurations 3 and 4.

## Material Control/Accounting of Special Nuclear Material (including US-IAEA Agreement

### *Background*

Part 70, Sections 51 and 63 provide general material balance, inventory, recordkeeping, and status report requirements that are applicable to nuclear power reactors. Section 53 refers to 10CFR74.13(a) and 75.35 which provide additional detailed material status report requirements including reporting form numbers and submittal dates.

Independent spent fuel storage installations have similar requirements as specified in 10CFR72.72, 72.76, and 75.35.

*Assessment*

The material control and accounting requirements of Parts 70, 74, and 75 remain fully applicable for permanently shutdown plants in spent fuel storage Configurations 1 and 2. Licensees in Configurations 3 should be exempted from the Section 70.51 and .53 and, as applicable, Part 74. Material accounting requirements will remain for the ISFSI under Parts 72 and, as applicable, Part 75. If all fuel is removed from the site, the material control and accounting requirements of Part 70, and all of Parts 74 and 75 are not longer applicable.

## Financial Protection Requirements

*Background*

The financial protection requirements for large nuclear power plants* are found in Part 140 of 10CFR. At the present time, paragraph 140.11(a)(4) requires a primary layer of financial protection of $200 million. A secondary layer of financial protection is also mandated. This is an industry retrospective rating plan providing for deferred premium charges equal to the pro rata share of the public liability claims and costs. Under this plan, the current maximum deferred premium charges for each nuclear reactor which is licensed to operate is $75.5 million with respect to any nuclear incident.** No more than $10 million per incident is required in a calendar year. The total financial protection for any incident would equal the primary layer of $200 million plus the secondary layer of $75.5 million times the number of reactors covered, or in excess of $8 billion.

This liability insurance covers claims resulting from a nuclear incident or a precautionary evacuation. In addition to accidents involving offsite releases, public evacuation and land contamination, the insurance covers liability arising from power plant effluents, storage and transportation of spent fuel,*** and radioactive waste materials. Included in the insurance coverage are defense costs for claims settlement.

10 CFR Part 140 was established in 1957 pursuant to Section 170 of the Atomic Energy Act of 1954, commonly called the Price-Anderson Act. One of the purposes of the Act was to protect the public by assuring the availability of funds for the payment of claims arising from a catastrophic nuclear incident. The Act required the AEC's reactor licensees to furnish financial protection (in the form of nuclear

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* i.e., a nuclear reactor facility that is designed for producing 100,000 electrical kilowatts or more.

** plus any surcharge assessed under subsection 170o (1)(E) of the Atomic Energy Act of 1954, as amended.

*** The liabilities and indemnification requirements associated with the transfer of spent fuel from the licensee to the Department of Energy will be evaluated on a case by case basis at a future time when spent fuel is shipped to a repository.

liability insurance or the equivalent) to cover public liability claims against the licensee and all others who might be liable for a nuclear incident. A second major provision required the AEC to indemnify the licensee and all others who might be liable in the amount of $500 million over and above the financial protection required. The Act also limited the liability from a nuclear incident to the sum of the financial protection required plus the AEC's indemnity. For large reactor licensees this resulted in a statutory liability limit of $560 million. The Act had similar provisions for certain licensees not operating reactors and to certain AEC contractors.

The financial protection requirement for large nuclear power plants was (and remains) the maximum amount of liability insurance available at a reasonable cost and on reasonable terms from private sources. The amount was originally $60 million. The required amount has been increased in step with increases in the amount of privately available nuclear energy liability insurance. The current requirement for this primary layer of insurance is $200 million. Other licensees generally have lesser financial protection requirements which consider type, size, and location of the licensed activity and "other factors pertaining to the hazard."

In 1975, the Price-Anderson Act was modified and extended until 1987 (Public Law 94-197). This amendment established a secondary layer of insurance by requiring that a retrospective premium of $2 to $5 million be established for large nuclear power plants. Part 140 was revised (42FR 46 1/3/77) to establish a retrospective premium of $5 million per facility per incident. The NRC chose the $5 million level because such a premium would not present an undue burden on any size utility. Moreover, since the $5 million requirement was the highest allowed by Public Law 94-197, it would result in the maximum financial protection available to pay public liability claims.

In 1988, Public Law 100-406 modified and extended the Price-Anderson Act to the year 2002. The retrospective premium was increased to $63 million per reactor per incident. This limit was subsequently increased to $75.5 million (58FR 42851 8/12/93) by Section t of the Act, based on the consumer price index change since 1988.

This discussion of the offsite liability insurance requirement has established that one intent of the Price-Anderson legislation is to protect the public by ensuring that timely compensation is available in the event of claims arising from a catastrophic nuclear incident. Unlike the onsite property damage insurance requirement, the offsite liability levels as mandated by Congress do not appear to have an explicit technical basis.

The primary insurance requirement, presently at $200 million, is based on the maximum amount of liability insurance available from private sources. Similarly, there does not appear to be an explicit technical basis for the secondary layer retrospective premium of $75.5 million per reactor.

Although the permanently shutdown nuclear power plant has a lower public risk, many activities that have the potential for public liability claims will continue until all radioactive materials are removed and the

site is released for unrestricted access. This implies that the offsite liability insurance requirement should continue although, for most configurations a lower requirement should suffice.

*Assessment*

There are three major considerations that are germane to this offsite liability assessment. Each is discussed below:

- **The Relationship of Accident Probability to the Liability Insurance Requirements**

One purpose of the Price-Anderson Act was to protect the public by assuring the availability of funds for the payment of claims arising from a catastrophic nuclear incident. Probabilistic Risk Assessments (PRAs) provide a mechanism to examine the relationship of accident frequency and accident consequence for a given enterprise. Full power PRAs of nuclear power plants show increasing consequences with decreasing accident frequencies. The accident consequences can be used to determine liability insurance levels.

Although Congress did not explicitly state its intent when specifying or amending the Price-Anderson Act, some inferences can be drawn from a review of the hearing transcripts.

On March 3, 1976, shortly after the Price-Anderson Amendments Act of 1976 (Public Law 94-197) was adopted, the Joint Committee on Atomic Energy held a hearing to consider whether the financial risk to utilities under the Price-Anderson system should be increased.[10] The hearing transcript provided the following insights:

From the prepared statement of Larry Hobart, Assistant General Manager, American Public Power Association (p. 34)

> *Public Law 94-197 was the result of extensive committee hearings and vigorous Congressional debate extending over a two-year period. During Congressional consideration of the legislation, the level of financial risk to be imposed on electric utilities was the major focus of attention. Testimony was taken on a variety of approaches to the question. The range of retrospective premiums provided under current law is the end-product of that very detailed examination.*
>
> *The decision by Congress took into account the conclusions of this committee relative to risk to the public, including evaluation of the findings of the study "An Assessment Accident Risks in U.S. Commercial Nuclear Power Plants" prepared under the direction of Dr. Norman C. Rasmussen of the Massachusetts Institute of Technology. The committee stated in its report of November 13, 1975, on this legislation that: "Insofar as the amount of financial protection for the public is concerned, both Dr. Rasmussen testimony before the joint Committee last year and the final report*

*affirm that the total of public and private indemnity provided for by this bill is adequate to cover any credible accident which might occur."*

As part of the general discussion, committee member representative John B. Anderson of Illinois stated (P. 11):

*One further comment on the question of the $560 million limit on liability. We did have some testimony before Joint Committee when we considered the extension of Price-Anderson to the effect that this would afford protection for about 96 percent of all the accidents that might occur.*

*In other words, that 96 percent of the probable accidents that could occur would be below the extent of the limits imposed on liability under this statute and the kind of accident that would exceed that amount would be one that would probably occur once in every 5,000 years and that as the pool floats upward, as it will do under the legislation, as I know the Senator is aware, to about $1 billion by 1985 this would include 99 percent of all accidents that might occur. In other words, accidents that would exceed that $1 billion would likely occur once in 10,000 years.*

The witness, Senator Charles H. Percy from Illinois responded in part,

*The committee was very wise to establish through the Rasmussen report the fact that the risks are relatively low. We needed some means of bringing it down from a 10,000-year span to what we can really comprehend in relation to our own insurance policies. We don't have to be concerned about 10,000 years so much as the probability of an accident occurring once in 10,000 chances in 1 year or once in a thousand chances in 10 years. The Rasmussen study shows that when 100 reactors are on line, the probability over a 10-year period of an accident with $900 million in property damages, a 2,000 square mile decontamination area, a 130 square mile relocation area, 300 early illnesses and total health effects over a 30 years of 5,100 latent cancer deaths, 42,000 thyroid nodules and several hundred genetic defects, is one in a thousand.*

On the basis of this testimony we can extrapolate that the frequency (F) of a release resulting in the stated consequences is:

F/ reactor year x 100 reactors x 10 years = 1.0E-3, therefore:

F = 1.0E-6/reactor year

These statements (and the intent of the Joint Committee) can be interpreted two ways:

1. The intent of the committee was to ensure that the primary and secondary layers of financial indemnity will afford protection for about 96 to 99 percent of the accidents that might occur.

2. The intent of the committee was to determine a credible accident frequency, and establish indemnity levels based on the estimate consequences of that credible accident.

For the purposes of operating power reactors these two interpretations have the same outcome. However, for the PSD plant they can produce disparate results, when the release frequency distribution is different from the full power operation of a nuclear plant. For example, if a release frequency ranged between 1E-7 and 1E-10, with 1E-9 and greater comprising 99 percent of the total frequency, interpretation number 1, would require the financial protection levels based on a 1E-9 accident. However, interpretation number 2 would not require any liability insurance.

It is likely that Congress implicitly assumed a credible accident frequency (interpretation number 2). We believe that the intent of Congress in establishing a retrospective premium in the range of $2 to $5 million was to ensure that adequate funds were available to cover any credible accident that might occur. That level of funds appears to be $1 billion. The associated "credible" accident frequency is about 1E-6 per reactor year.

The release frequency estimates for the spent fuel storage configuration representative accident sequences are provided in Section 3. The release frequency for the Configuration 1 accident is in the E-6 range for both BWRs and PWRs. The spent fuel assembly drop (Configuration 2) is 3E-4 events per year. The ISFSI release frequency (Configuration 3) of 6E-6 events per year is from an EPRI study. However, as discussed in Section 3, it is our judgement that this frequency is overstated by at least two orders of magnitude. The estimated release frequency is approximately 3E-7 events per year. The Configuration 4 seismically induced borated water storage tank (BWST) rupture has been estimated at 2E-7 events per year.

**Table B.2 A Comparison of Consequence Estimates**

| | Early Fatalities | Latent Fatalities | Societal Dose (Person-Rem) | Boundary Dose (Rem) | Condemned Land (Sq. Miles) | Total Offsite Costs $ |
|---|---|---|---|---|---|---|
| Configuration 2 | 0 | 2 | 4000 | 0.009 | 0.0 | neg |
| Configuration 3 | 0 | 0.22 | 690 | 0.472 | 0.0002 | neg |
| TMI 2[1] | 0 | 0.4 | ~2000 | 0.100 | 0.00 | neg[2] |

1. TMI 2 accident information is from the Rogovin Report (Ref. 12)
2. Established based on milk and vegation sampling results reported in Reference 12. All samples were well under EPA protective action levels.

- **The Relationship that Accident Consequence Calculations Have to Actual Liability Expenses**

Consequence codes such as the MELCOR Accident Consequence Code System (MAACS) are used to estimate the outcomes of radiological accidents in terms of health effects, population dose, and economic cost. It appears that one bases of the offsite liability requirement for large power reactors is an estimate of accident consequences. However, these calculations are not necessarily representative of actual experience.

For example, Table B.2 presents the consequence estimates for Configurations 2 and 3 using the MAACS Code. The Three Mile Island Unit 2 accident data is also provided for comparison. The table shows that the TMI 2 offsite health and economic consequences are similar to the estimates for Configuration 2 and 3. Yet, as of 1993, $60 million has been awarded settlement of claims arising from the TMI 2 accident. A significant number of claims were still unsettled as of 1993.[11]

There clearly is a disparity between the expected consequences and the public's perception of an accident. The Rogovin Report[12] recognized this stating:

> *In our view, the fact that there will be no adverse radiation health effects, or very minimal effects, from the Three Mile Island accident has not been clearly understood by the public. It is clear to us that the public misconception about the risks associated with the actual releases measured during the accident, as well as about the risks associated with nuclear power plants generally, has been due to a failure to convey credible information regarding the actual risks in an understandable fashion to the public.*

Despite significant education efforts, the majority of the public is not comfortable with nuclear power. In all likelihood, the public mistrust of all things nuclear will continue for the foreseeable future. In this environment the public reaction to relatively minor incidents will be exacerbated, (e.g., precautionary evacuation) and result in economic consequences that are far in excess of code predictions.

- **The PriceAnderson Requirements for Non Operating Reactors and ISFSIs**

Section 170 of the Atomic Energy Act,* Part a requires that:

> *"Each licensee issued under Section 103 or 104 and each construction permit issued under Section 185 shall, and each licensee issued under Section 53, 63, or 81* ***may****, for the public purposes cited in Section 2 I. have as a condition of the license a requirement that the licensee have and maintain financial protection of such type and in such amounts as the Nuclear Regulatory Commission (in this*

*Commonly known as the Price-Anderson Act.

*section referred to as the "Commission") in the exercise of its licensing and regulatory authority and responsibility shall require..." (emphasis added)*

The NRC must require financial protection for licensees issued under Section 103 (commercial licenses), Section 104 (medical therapy and research and development) and for construction permits and operating licenses under Section 185. Section 170b gives the Commission the authority to require less than the maximum amount of primary financial protection, in consideration of other factors including, the type, size, and locations of the licensed activity. However, the Act specifies primary and secondary insurance amounts for facilities designed for producing substantial amount of electricity. Financial protection is not mandated for Sections 53, 63, and 81 which addresses the domestic distribution of: special nuclear material, source material, and byproduct material, respectively.

There has been significant debate regarding the applicability of Section 170 to permanently shutdown facilities. After a sufficient cooling period such that there is no longer the threat of rapid zircaloy oxidation, the accidents that could be associated with the PSD facility have significantly reduced consequences. Cases can be made for removing the offsite liability insurance requirement or continuing it with less than the maximum amount required for the permanently shutdown facility.

Section 4 of this report developed accident consequence estimates for the four spent fuel storage configurations that were assessed for this program.

Configuration 1, "Hot Fuel in the Spent Fuel Pool," postulated rapid zircaloy oxidation of the spent fuel rods after the loss of the pool water inventory. The safety hazard analysis has estimated consequences that are approximately equal to a severe core damage accident.

Configuration 2, "Cold Fuel in the Spent Fuel Pool," has sufficiently low decay heat loads such that the cladding will remain intact even if all spent fuel pool water is lost. Configuration 2 considers the consequences of a dropped assembly. The safety hazard analysis, as discussed in Section 4 of shows negligible offsite costs.

In lieu of long term storage in the spent fuel pool, a permanently shutdown nuclear power plant may store its spent fuel in an Independent Spent Fuel Storage Installation (ISFSI), before, during, and after, the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. The postulated accident for Configuration 3 is a breach of the ISFSI which damages a single BWR or PWR fuel assembly.* The estimated offsite cost is negligible

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*This consequence estimate may not envelope sabotage scenarios which could conceivably involve a greater radionuclide release. These scenarios are safeguard information. The information on radionuclide release (if any) is not available to BNL.

Configuration 4, "All Fuel Removed from the Site," assumes that all spent fuel has been shipped offsite, including any that might have been stored in an ISFSI. As discussed in Section 4, the postulated accidental radioactive releases to the atmosphere during decommissioning do not pose a significant threat to the onsite workers or the public. For the purpose of estimating onsite accident cleanup costs, the postulated scenario for Configuration 4 is the rupture of the borated water storage tank. Approximately 450,000 gallons of slightly radioactive water is released causing onsite soil contamination and potential contamination of the water table. BNL has performed calculations that indicate tritium levels will be below the maximum concentration limit for drinking water at the site boundary. Offsite remediation has not been considered and again offsite costs are considered to be negligible.

Given the potential magnitude of the consequences, it is appropriate that the offsite liability insurance requirements of 10CFR Part 140, both the primary and secondary levels, remain in place for Configuration 1.

The insurance recommendations for the remaining configurations are not as straightforward. Qualitative justifications can be made for anywhere from $0 to $200 million.

Since the analyses show minimal offsite consequences, a case can be made for eliminating the offsite liability requirements for Configurations 2, 3, and 4. Any liability awards *should* be minimal and the licensee should be able to pay those awards in a timely manner, thereby satisfying the intent of the Price-Anderson Act.

Conversely, the $200 million figure recognizes the possibility of a large suit for alleged damages due to routine, low level radioactive effluents from the plant during decommissioning.

All things considered, a $100 million offsite liability insurance requirement is a reasonable compromise for the permanently shutdown plant. The TMI 2 experience has shown that significant judgements can be awarded, despite negligible offsite consequences. It is also recommended that these plants be allowed to withdraw from the secondary level of protection. In addition, the exemption process could be used to justify lower plant specific requirements, as deemed appropriate.

For the independent spent fuel storage installations (ISFSIs) that are not covered under an existing site policy, it is acknowledged that a lower liability limit could be justified. The passive nature of the installation, and the expected lack of radioactive effluents, routine or otherwise, conceivably results in less liability exposure.

## Annual Fees for Licensees

### *Background*

Part 171 of 10CFR, "Annual Fees for Nuclear Power Reactor Operating Licenses," was published on September 18, 1986 [51FR33224] as a final rule. The rule assessed an annual fee for FY1987 for every power reactor licensed to operate. The annual fee was instituted to comply with the statutory mandate of the Consolidated Omnibus Budget Reconciliation Act of 1985. The scope of this section was expanded [56FR31472, 7/10/91] to include other entities including nonpower reactors, materials licensees, part 72 ISFSI licensees, fuel facilities, etc., in response to the congressional mandate requiring the NRC to recover approximately 100% of its budget authority in FY1991 and the four succeeding years. In the Responses to Comments, Section D, Specific Fee Issues of the Final Rule, the NRC responded to the issue of annual fees for shutdown plants. Two commenters had indicated that charging the full annual power reactor fee was not fair because certain costs allocated to all power reactors were not applicable to permanently shutdown plants. The Commission responded that the proposed rule excluded power reactors with a POL* from the FY 1991 fee base. This waiver was extended and remains in effect for FY95.

### *Assessment*

The NRC is required to recover approximately 100% of its budget authority. The licensing and inspection fees assessed under Part 170 recover the costs of providing individually identifiable services to specific applicants for, and holders of, NRC licenses and approvals. Part 171 provides for the recovery of NRC budgeted costs for generic regulatory activities for each class of licensee. For example, the generic activities associated with power reactor licensees include: reactor decommissioning, license renewal, construction permit, and operating license reviews. Also included are generic costs such as the Incident Response Center and certain other NRC efforts that can support other licensees, but are primarily established for the power reactor licensee. Costs attributable to types of licenses other than power reactors (i.e., part 72 licensees) consists of generic regulatory costs and other costs not recoverable under Part 170, including rulemaking, upgrading safeguards requirements, modifying the standard review plans and developing inspection programs.

Permanently shutdown power reactor licensees continue to require NRC services, although not to the extent of a full power licensee. It is recommended that the Part 50 licensees, authorized to possess but not operate a nuclear power reactor be assessed as a group for the NRC services that are to be provided. If the appropriate fees cannot be accurately assessed at this time, perhaps a fee that is equivalent to the annual ISFSI fee can be instituted.

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*or with a formal NRC order prohibiting placing fuel back in the reactor vessel.

## B.4 REFERENCES


1. Title 10 of the Code of Federal Regulations, Parts 0-100, as of January 1, 1995.

2. Smith, R.I. et al., "Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," NUREG/CR-0130, June 1978.

3. Oak, H.D. et al., "Technology, Safety, andCosts of Decommissioning a Reference Boiling Water Reactor Power Station," NUREG/CR-0672, June 1980.

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An evaluation of the nuclear power plant regulatory basis is performed, as it pertains to those plants that are permanently shutdown (PSD) and awaiting or undergoing decommissioning. Four spent fuel storage configurations are examined. Recommendations are provided for those operationally based regulations that could be partially or totally removed for PSD plants without impacting public health and safety.

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# Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States

Robert Alvarez, Jan Beyea, Klaus Janberg, Jungmin Kang, Ed Lyman, Allison Macfarlane, Gordon Thompson, Frank N. von Hippel

Because of the unavailability of off-site storage for spent power-reactor fuel, the NRC has allowed high-density storage of spent fuel in pools originally designed to hold much smaller inventories. As a result, virtually all U.S. spent-fuel pools have been re-racked to hold spent-fuel assemblies at densities that approach those in reactor cores. In order to prevent the spent fuel from going critical, the fuel assemblies are partitioned off from each other in metal boxes whose walls contain neutron-absorbing boron. It has been known for more than two decades that, in case of a loss of water in the pool, convective air cooling would be relatively ineffective in such a "dense-packed" pool. Spent fuel recently discharged from a reactor could heat up relatively rapidly to temperatures at which the zircaloy fuel cladding could catch fire and the fuel's volatile fission products,

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**Robert Alvarez**, Senior Scholar, Institute for Policy Studies, 733 15 St NW, Suite 1025, Washington, DC. E-mail: kitbob@erols.com
**Jan Beyea**, Consulting in the Public Interest, 53 Clinton St, Lambertville, NJ 08530. E-mail: jbeyea@cipi.com
**Klaus Janberg**, 40 J.P. Melchiorstr., 40996, Ratingen, Germany. E-mail: klaus@janberg.de
**Jungmin Kang**, 101-1005 Chunggu Apt., Jamwon-dong, Seocho-gu, Seoul, South Korea. E-mail: jmkang55@hotmail.com
**Ed Lyman**, Nuclear Control Institute, Suite 410, 1000 Conn. Ave, NW, Washington, DC 20036. E-mail: lyman@nci.org
**Allison Macfarlane**, Security Studies Program, Center for International Studies, Massachusetts Institute of Technology, E38-620, 292 Main Street, Cambridge, MA 02139. E-mail: allisonm@mit.edu
**Gordon Thompson**, Institute for Resource and Security Studies, 27 Ellsworth Avenue, Cambridge, MA 02139. E-mail: irss@igc.org
**Frank N. von Hippel**, Program on Science and Global Security, 221 Nassau St, 2nd Floor, Princeton University, Princeton, NJ 08542-4601. E-mail: fvhippel@princeton.edu

including 30-year half-life $^{137}$Cs, would be released. The fire could well spread to older spent fuel. The long-term land-contamination consequences of such an event could be significantly worse than those from Chernobyl.

No such event has occurred thus far. However, the consequences would affect such a large area that alternatives to dense-pack storage must be examined—especially in the context of concerns that terrorists might find nuclear facilities attractive targets. To reduce both the consequences and probability of a spent-fuel-pool fire, it is proposed that all spent fuel be transferred from wet to dry storage within five years of discharge. The cost of on-site dry-cask storage for an additional 35,000 tons of older spent fuel is estimated at $3.5–7 billion dollars or 0.03–0.06 cents per kilowatt-hour generated from that fuel. Later cost savings could offset some of this cost when the fuel is shipped off site. The transfer to dry storage could be accomplished within a decade. The removal of the older fuel would reduce the average inventory of $^{137}$Cs in the pools by about a factor of four, bringing it down to about twice that in a reactor core. It would also make possible a return to open-rack storage for the remaining more recently discharged fuel. If accompanied by the installation of large emergency doors or blowers to provide large-scale airflow through the buildings housing the pools, natural convection air cooling of this spent fuel should be possible if airflow has not been blocked by collapse of the building or other cause. Other possible risk-reduction measures are also discussed.

Our purpose in writing this article is to make this problem accessible to a broader audience than has been considering it, with the goal of encouraging further public discussion and analysis. More detailed technical discussions of scenarios that could result in loss-of-coolant from spent-fuel pools and of the likelihood of spent-fuel fires resulting are available in published reports prepared for the NRC over the past two decades. Although it may be necessary to keep some specific vulnerabilities confidential, we believe that a generic discussion of the type presented here can and must be made available so that interested experts and the concerned public can hold the NRC, nuclear-power-plant operators, and independent policy analysts such as ourselves accountable.

## INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has estimated the probability of a loss of coolant from a spent-fuel storage pool to be so small (about $10^{-6}$ per pool-year) that design requirements to mitigate the consequences have not been required.[1] As a result, the NRC continues to permit pools to move from open-rack configurations, for which natural-convection air cooling would have been effective, to "dense-pack" configurations that eventually fill pools almost wall to wall. A 1979 study done for the NRC by the Sandia National Laboratory showed that, in case of a sudden loss of all the water in a pool, dense-packed spent fuel, even a year after discharge, would likely heat up to the point where its zircaloy cladding would burst and then catch fire.[2] This would result in the airborne release of massive quantities of fission products.

No such event has occurred thus far. However, the consequences would be so severe that alternatives to dense-pack storage must be examined—especially

in the context of heightened concerns that terrorists could find nuclear facilities attractive targets.

The NRC's standard approach to estimating the probabilities of nuclear accidents has been to rely on fault-tree analysis. This involves quantitative estimates of the probability of release scenarios due to sequences of equipment failure, human error, and acts of nature. However, as the NRC staff stated in a June 2001 briefing on risks from stored spent nuclear fuel:[3] "No established method exists for quantitatively estimating the likelihood of a sabotage event at a nuclear facility."

Recently, the NRC has denied petitions by citizen groups seeking enhanced protections from terrorist acts against reactor spent-fuel pools.[4] In its decision, the NRC has asserted that "the possibility of a terrorist attack . . . is speculative and simply too far removed from the natural or expected consequences of agency action . . . "[5]

In support of its decision, the NRC stated: "Congress has recognized the need for and encouraged high-density spent fuel storage at reactor sites,"[6] referencing the 1982 Nuclear Waste Policy Act (NWPA). In fact, although the NWPA cites the need for "the effective use of existing storage facilities, and necessary additional storage, at the site of each civilian nuclear power reactor consistent with public health and safety," it does *not* explicitly endorse dense-pack storage.[7]

If probabilistic analysis is of little help for evaluating the risks of terrorism, the NRC and the U.S. Congress will have to make a judgment of the probability estimates that will be used in cost-benefit analyses. Here, we propose physical changes to spent-fuel storage arrangements that would correct the most obvious vulnerabilities of pools to loss of coolant and fire. The most costly of these proposals, shifting fuel to dry cask storage about 5 years after discharge from a reactor, would cost $3.5–7 billion for dry storage of the approximately 35,000 tons of older spent fuel that would otherwise be stored in U.S. pools in 2010. This corresponds to about 0.03–0.06 cents per kilowatt-hour of electricity generated from the fuel. Some of this cost could be recovered later if it reduced costs for the shipment of the spent fuel off-site to a long-term or permanent storage site.

For comparison, the property losses from the deposition downwind of the cesium-137 released by a spent-fuel-pool fire would likely be hundreds of billions of dollars. The removal of the older spent fuel to dry storage would therefore be justified by a traditional cost-benefit analysis if the likelihood of a spent-fuel-pool fire in the U.S. during the next 30 years were judged to be greater than about a percent. Other actions recommended below could be justified by much lower probabilities.

It appears unlikely that the NRC will decide its own to require such actions. According to its Inspector General, the "NRC appears to have informally established an unreasonably high burden of requiring absolute proof of a safety problem, versus lack of a reasonable assurance of maintaining public health and safety . . . "[8]

This situation calls for more explicit guidance from Congress. Indeed, 27 state Attorneys General have recently signed a letter to Congressional leaders asking for legislation to "protect our states and communities from terrorist attacks against civilian nuclear power plants and other sensitive nuclear facilities," specifically mentioning spent-fuel pools.[9]

Congress could do this by updating the Nuclear Waste Policy Act to require "defense in depth" for pool storage; and the minimization of pool inventories of spent fuel. The second requirement would involve the transfer, over a transition period of not more than a decade, of all spent fuel more than five years post discharge to dry, hardened storage modes.

To establish the basis for an informed, democratic decision on risk-reduction measures, it would be desirable to have the relevant analysis available to a full range of concerned parties, including state and local governments and concerned citizens. Despite the need to keep sensitive details confidential, we believe that we have demonstrated in this article that analysts can describe and debate a range of measures in an open process. The same can be done in the regulatory area. Evidentiary hearings held under NRC rules already have specific provisions to exclude security details—along with proprietary and confidential personnel information—from the public record.

In outline, we describe:

- The huge inventories of the long-lived, volatile fission product cesium-137 ($^{137}Cs$) that are accumulating in U.S. spent fuel pools and the consequences if the inventory of one of these pools were released to the atmosphere as a result of a spent-fuel fire;
- The various types of events that have been discussed in the public record that could cause a loss of coolant and the high radiation levels that would result in the building above the pool as a result of the loss of the radiation shielding provided by the water;
- The limitations of the various cooling mechanisms for dry spent fuel: conduction, infra-red radiation, steam cooling and convective air cooling;
- Possible measures to reduce the vulnerability of pools to a loss of coolant event and to provide emergency cooling if such an event should occur; and

- The feasibility of moving spent fuel from pools into dry-cask storage within 5 years after discharge from the reactor. This would allow open-rack storage of the more-recently discharged fuel, which would make convective air-cooling more effective in case of a loss of water, and would reduce the average inventory of $^{137}Cs$ in U.S. spent-fuel pools by about a factor of four.

There are 103 commercial nuclear reactors operating in the U.S. at 65 sites in 31 states (Figure 1).[11] Of these, 69 are pressurized-water reactors (PWRs) and 34 are boiling-water reactors (BWRs). In addition there are 14 previously-operating light-water-cooled power reactors in various stages of decommissioning. Some of these reactors share spent-fuel pools, so that there is a total of 65 PWR and 34 BWR pools.[12] Figure 2 shows diagrams of "generic" pressurized-water reactor (PWR) and boiling-water-reactor (BWR) spent-fuel pools.[13] For simplicity, when we do illustrative calculations in this article, we use PWR fuel and pool designs. However, the results of detailed studies done for the NRC show that our qualitative conclusions are applicable to BWRs as well.[14]

**Figure 1:** Locations of nuclear power plants in the United States. Circles represent sites with one reactor, squares represent plants with two; and stars represent plants with three. Open symbols represent sites with at least one shutdown reactor. Only the plant in Zion, Illinois has more than one shutdown reactor. It has two (*Source:* authors[10]).

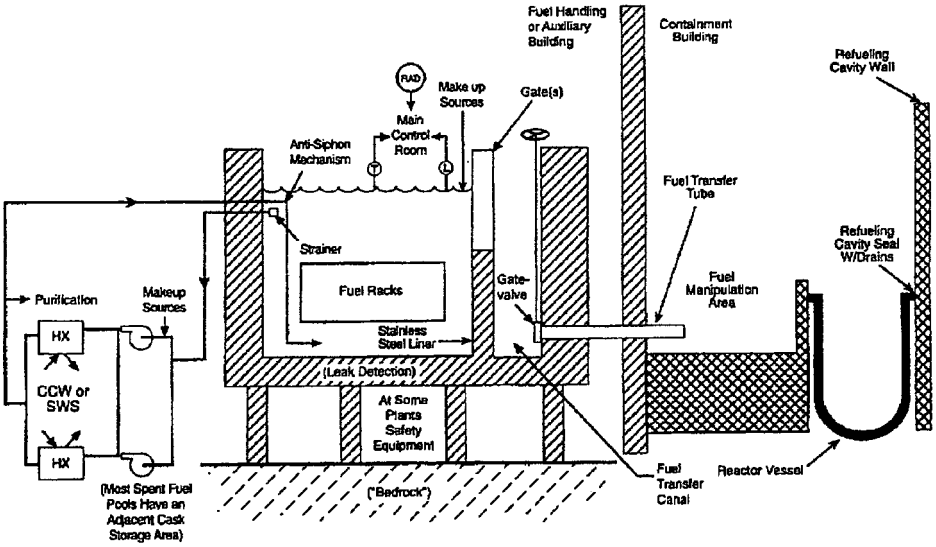



**Figure 2a:** Layout of spent fuel pool and transfer system for pressurized water reactors (*Source:* NUREG-1275, 1997).

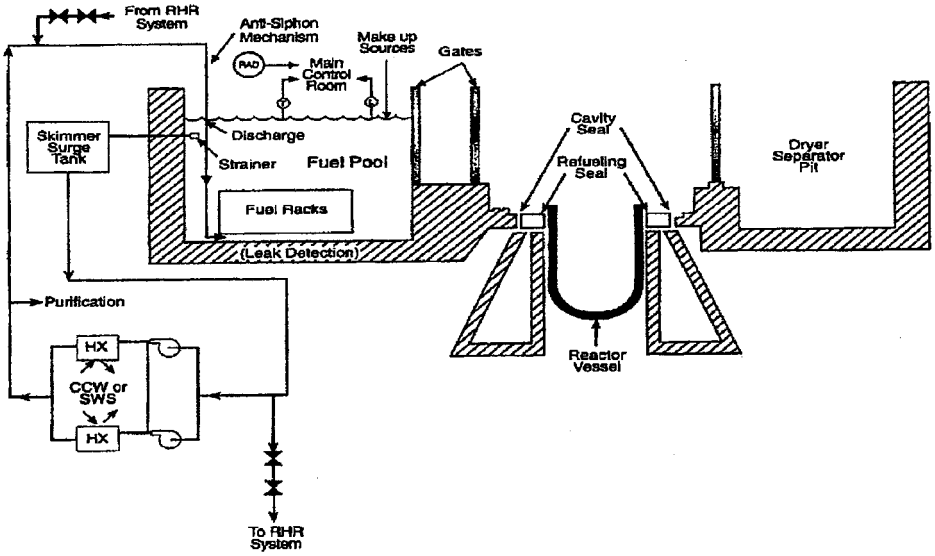



**Figure 2b:** Layout of spent fuel pool and transfer system for boiling water reactors (*Source:* NUREG-1275, 1997).

## THE HAZARD FROM CESIUM-137 RELEASES

Although a number of isotopes are of concern, we focus here on the fission product $^{137}$Cs. It has a 30-year half-life, is relatively volatile and, along with its short-lived decay product, barium-137 (2.55 minute half-life), accounts for about half of the fission-product activity in 10-year-old spent fuel.[15] It is a potent land contaminant because 95% of its decays are to an excited state of $^{137}$Ba, which de-excites by emitting a penetrating (0.66-MeV) gamma ray.[16]

The damage that can be done by a large release of fission products was demonstrated by the April 1986 Chernobyl accident. More than 100,000 residents from 187 settlements were permanently evacuated because of contamination by $^{137}$Cs. Strict radiation-dose control measures were imposed in areas contaminated to levels greater than 15 Ci/km$^2$ (555 kBq/m$^2$) of $^{137}$Cs. The total area of this radiation-control zone is huge: 10,000 km$^2$, equal to half the area of the State of New Jersey. During the following decade, the population of this area declined by almost half because of migration to areas of lower contamination.[17]

*Inventories of Cs-137 in Spent-Fuel Storage Pools*

The spent-fuel pools adjacent to most power reactors contain much larger inventories of $^{137}$Cs than the 2 MegaCuries (MCi) that were released from the core of Chernobyl 1000-Megawatt electric (MWe) unit #4[18] or the approximately 5 MCi in the core of a 1000-MWe light-water reactor. A typical 1000-MWe pressurized water reactor (PWR) core contains about 80 metric tons of uranium in its fuel, while a typical U.S. spent fuel pool today contains about 400 tons of spent fuel (see Figure 3). (In this article, wherever tons are referred to, metric tons are meant.) Furthermore, since the concentration of $^{137}$Cs builds up almost linearly with burnup, there is on average about twice as much in a ton of spent fuel as in a ton of fuel in the reactor core.

For an average cumulative fission energy release of 40 Megawatt-days thermal per kg of uranium originally in the fuel (MWt-days/kgU) and an average subsequent decay time of 15 years, 400 tons of spent power-reactor fuel would contain 35 megaCuries (MCi) of $^{137}$Cs.[19] If 10–100% of the $^{137}$Cs in a spent-fuel pool,[20] i.e., 3.5–35 MCi, were released by a spent-fuel fire to the atmosphere in a plume distributed vertically uniformly through the atmosphere's lower "mixing layer" and dispersed downwind in a "wedge model" approximation under median conditions (mixing layer thickness of 1 km, wedge opening angle of 6 degrees, wind speed of 5 m/sec, and deposition velocity of 1 cm/sec) then 37,000–150,000 km$^2$ would be contaminated above 15 Ci/km$^2$, 6,000–50,000 km$^2$ would

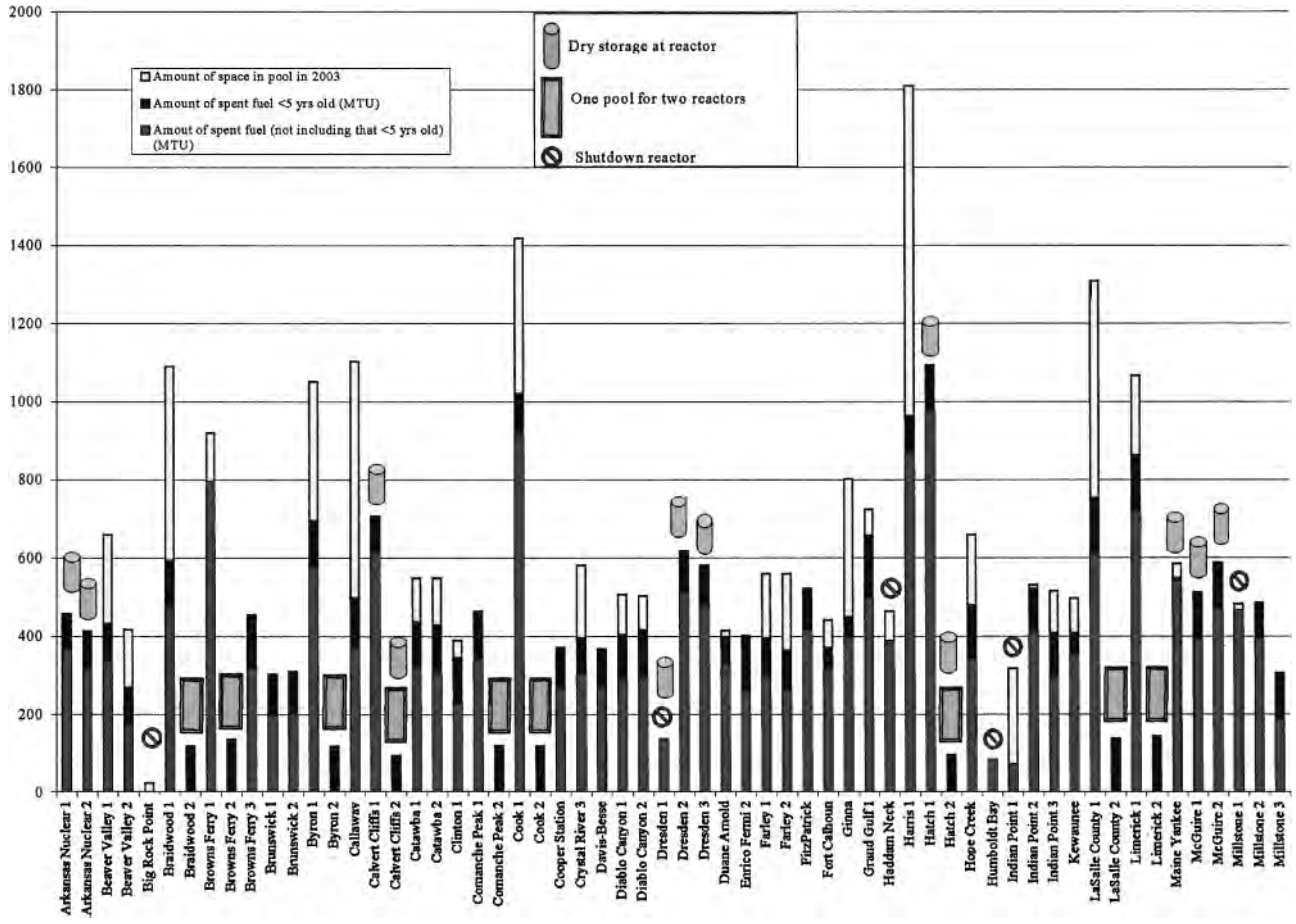



**Figure 3:** Estimated 2003 spent fuel inventory at each U.S. spent-fuel pool, measured in metric tons of contained uranium. Height of bar indicates total licensed capacity (1998, with some updates). Shading indicates estimated tonnage of spent fuel in pool as of 2003. Dark shading indicates the estimated amount of fuel discharged from the reactors within the past 5 years. Canister indicates the presence of on-site dry storage. Pool indicates that reactor shares a pool with the reactor to the left (*Source:* authors[25]). *(Continued)*

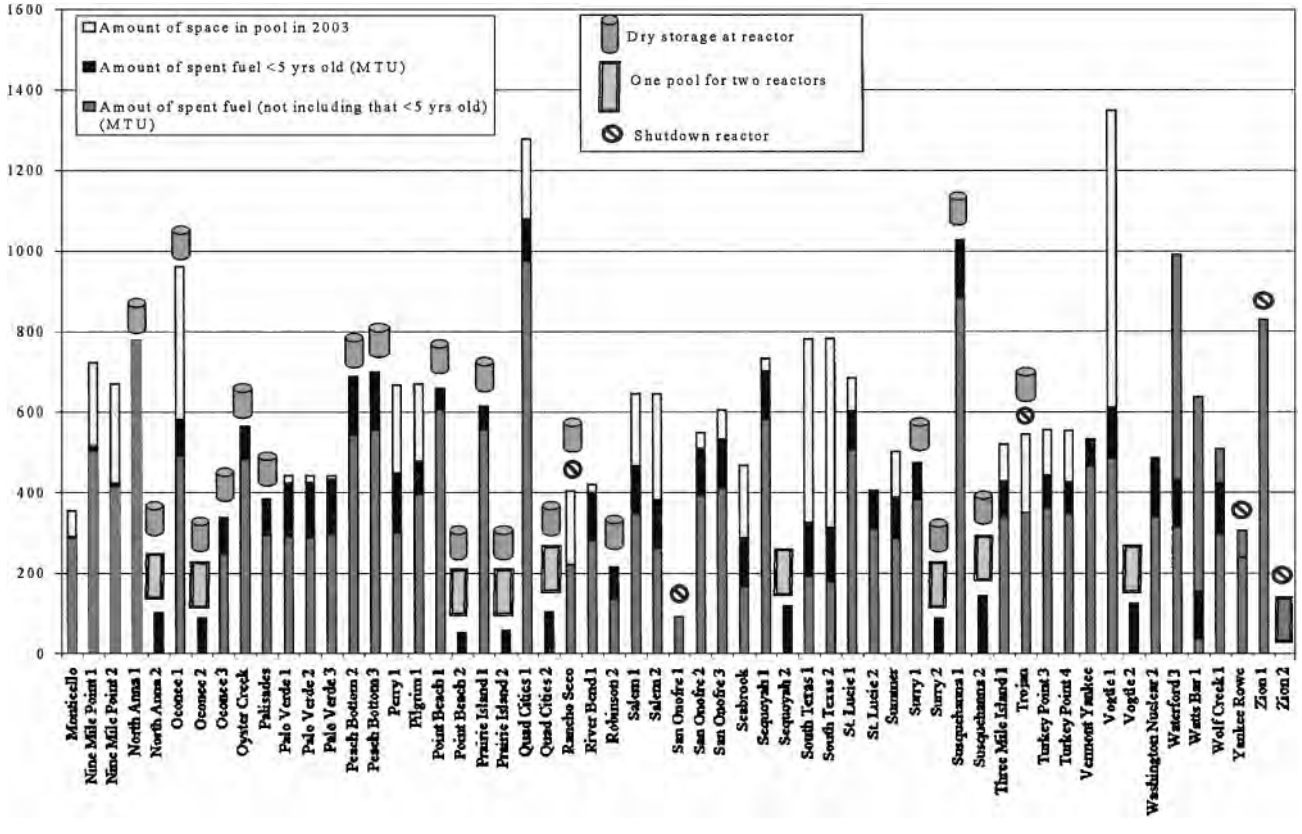



**Figure 3:** (Continued)

**Table 1:** Typical plume areas ($km^2$).

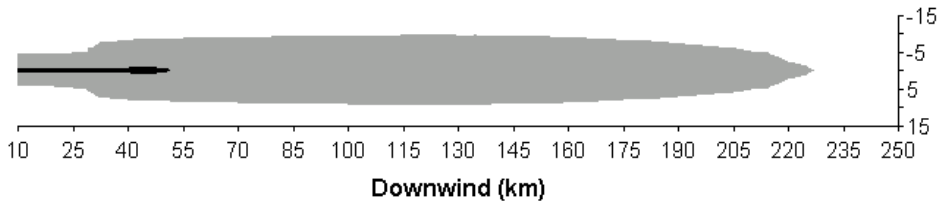
| Release | $>$100 Ci/$km^2$ | $>$1000 Ci/$km^2$ |
|---|---|---|
| Chernobyl (2 MCi, hot, multi-directional) | $\approx$700 | |
| 3.5 MCi (MACCS2) | 3,500 | 200 |
| 3.5 MCi (wedge model) | 6,000 | 180 |
| 35 MCi (MACCS2) | 45,000 | 2,500 |
| 35 MCi (wedge model) | 50,000 | 6,000 |

be contaminated to greater than 100 Ci/$km^2$ and 180–6000 $km^2$ to a level of greater than 1000 Ci/$km^2$.[21] Table 1 and Figure 4 show typical contaminated areas, calculated using the MACCS2 Gaussian plume dispersion code used by the NRC[22] for fires with 40 MWt thermal power.[23] This corresponds to fire durations of half an hour and 5 hours, respectively for fires that burn 10 or 100 percent of 400 tons of spent fuel.[24] Similar results were obtained for slower-burning fires with powers of 5 MWt.

It will be seen in Table 1 that, for the 3.5 MCi release, the area calculated as contaminated above 100 Ci/$km^2$ are 5–9 times larger than the area contaminated to this level by the 2 MCi release from the Chernobyl accident. The reasons are that, at Chernobyl: 1) much of the Cs-137 was lifted to heights of up to 2.5 km by the initial explosion and the subsequent hot fire and therefore carried far downwind;[26] and 2) the release extended over 10 days during which the wind blew in virtually all directions. As a result, more than 90 percent of the $^{137}$Cs from Chernobyl was dispersed into areas that were contaminated to less than 40 Ci/$km^2$.[27] In contrast, in the wedge-model calculations for the 3.5 MCi release, about 50 percent of the $^{137}$Cs is deposited in areas contaminated to greater than this level.

The projected whole-body dose from external radiation from $^{137}$Cs to someone living for 10 years in an area contaminated to 100 or 1000 Ci/$km^2$ would be 10–20 or 100–200 rem, with an associated additional risk of cancer death of about 1 or 10 percent respectively.[28] A 1 or 10 percent added risk would increase an average person's lifetime cancer death risk from about 20 percent to 21 or 30 percent.

A 1997 study done for the NRC estimated the median consequences of a spent-fuel fire at a pressurized water reactor (PWR) that released 8–80 MCi of $^{137}$Cs. The consequences included: 54,000–143,000 extra cancer deaths, 2000–7000 $km^2$ of agricultural land condemned, and economic costs due to evacuation of \$117–566 billion.[29] This is consistent with our own calculations using the MACCS2 code. It is obvious that all practical measures must be taken to prevent the occurrence of such an event.

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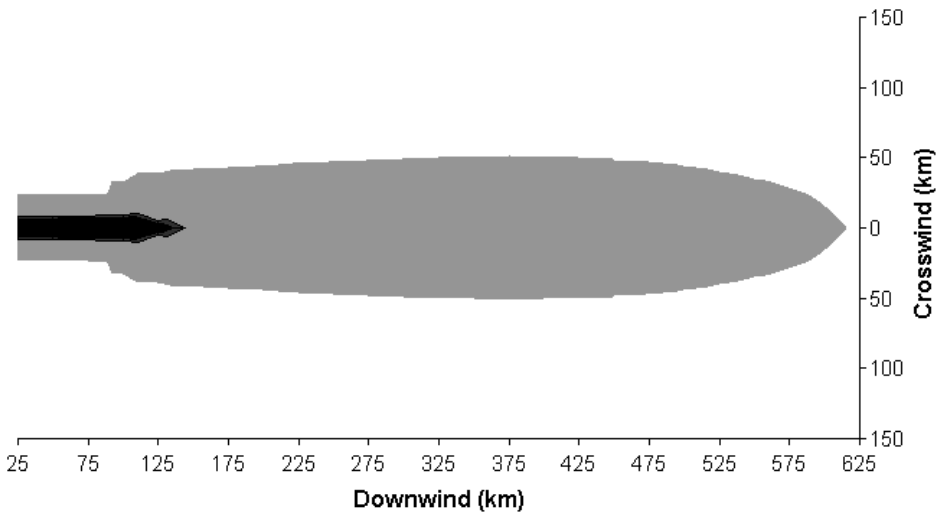


(b)

**Figure 4:** Typical areas contaminated above 100 (shaded) and 1000 (black) Ci/km$^2$ for release of (a) 3.5 MCi and (b) 35 MCi of $^{137}$Cs. The added chance of cancer death for a person living within the shaded area for 10 years is estimated very roughly as between 1 and 10 percent. For someone living within the black area, the added risk would be greater than 10 percent (i.e. the "normal" 20% lifetime cancer death risk would be increased to over 30 percent.) (*Source:* authors).

## SCENARIOS FOR A LOSS OF SPENT-FUEL-POOL WATER

The cooling water in a spent-fuel pool could be lost in a number of ways, through accidents or malicious acts. Detailed discussions of sensitive information are not necessary for our purposes. Below, we provide some perspective for the following generic cases: boil-off; drainage into other volumes through the opening of some combination of the valves, gates and pipes that hold the water in the pool; a fire resulting from the crash of a large aircraft; and puncture by an aircraft turbine shaft or a shaped charge.

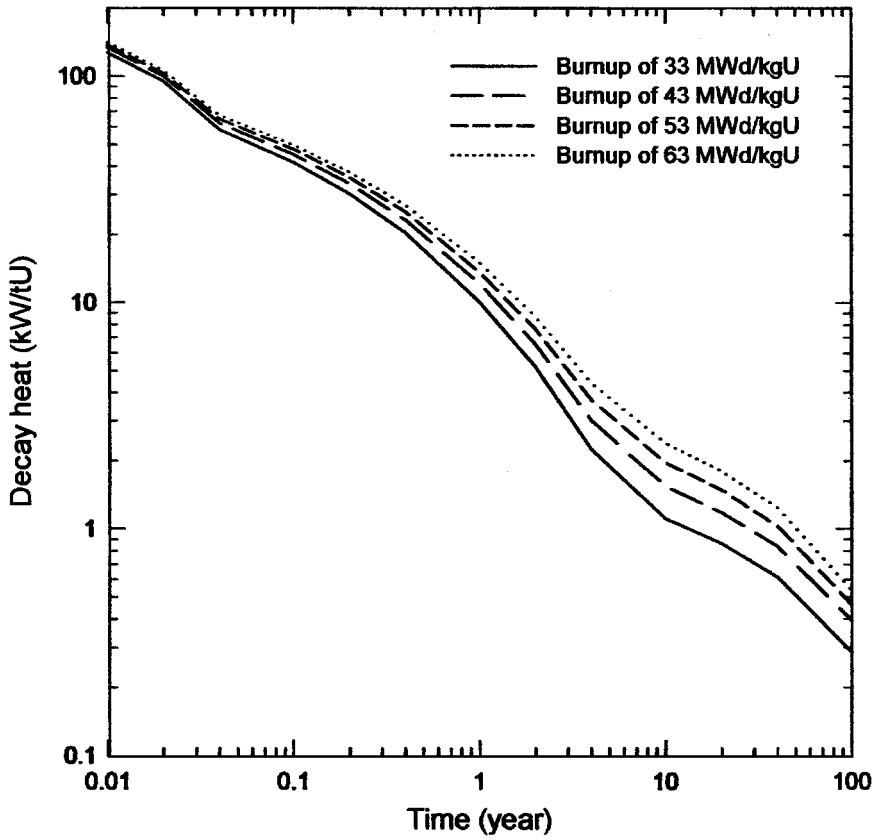


**Figure 5:** Decay heat as a function of time from 0.01 years (about 4 days) to 100 years for spent-fuel burnups of 33, 43, 53 and 63 MWd/kgU. The lowest burnup was typical for the 1970s. Current burnups are around 50 MWd/kgU (*Source:* authors[38]).

*Boil Off*

Keeping spent fuel cool is less demanding than keeping the core in an operating reactor cool. Five minutes after shutdown, nuclear fuel is still releasing 800 kilowatts of radioactive heat per metric ton of uranium (kWt/tU)[30]. However, after several days, the decay heat is down to 100 kWt/tU and after 5 years the level is down to 2–3 kWt/tU (see Figure 5).

In case of a loss of cooling, the time it would take for a spent-fuel pool to boil down to near the top of the spent fuel would be more than 10 days if the most recent spent-fuel discharge had been a year before. If the entire core of a reactor had been unloaded into the spent fuel pool only a few days after shutdown, the time could be as short as a day.[31] Early transfer of spent fuel into

storage pools has become common as reactor operators have reduced shutdown periods. Operators often transfer the entire core to the pool in order to expedite refueling or to facilitate inspection of the internals of the reactor pressure vessel and identification and replacement of fuel rods leaking fission products.[32]

Even a day would allow considerable time to provide emergency cooling if operators were not prevented from doing so by a major accident or terrorist act such as an attack on the associated reactor that released a large quantity of radioactivity. In this article, we do not discuss scenarios in which spent-fuel fires compound the consequences of radioactive releases from reactors. We therefore focus on the possibility of an accident or terrorist act that could rapidly drain a pool to a level below the top of the fuel.

*Drainage*

All spent-fuel pools are connected via fuel-transfer canals or tubes to the cavity holding the reactor pressure vessel. All can be partially drained through failure of interconnected piping systems, moveable gates, or seals designed to close the space between the pressure vessel and its surrounding reactor cavity.[33] A 1997 NRC report described two incidents of accidental partial drainage as follows:[34]

> Two loss of SFP [spent fuel pool] coolant inventory events occurred in which SFP level decrease exceeded 5 feet [1.5 m]. These events were terminated by operator action when approximately 20 feet [6 m] of coolant remained above the stored fuel. Without operator actions, the inventory loss could have continued until the SFP level had dropped to near the top of the stored fuel resulting in radiation fields that would have prevented access to the SFP area.

Once the pool water level is below the top of the fuel, the gamma radiation level would climb to 10,000 rems/hr at the edge of the pool and 100's of rems/hr in regions of the spent-fuel building out of direct sight of the fuel because of scattering of the gamma rays by air and the building structure (see Figure 6).[35] At the lower radiation level, lethal doses would be incurred within about an hour.[36] Given such dose rates, the NRC staff assumed that further *ad hoc* interventions would not be possible.[37]

*Fire*

A crash into the spent fuel pool by a large aircraft raises concerns of both puncture (see below) and fire. With regard to fire, researchers at the Sandia National Laboratory, using water to simulate kerosene, crashed loaded airplane

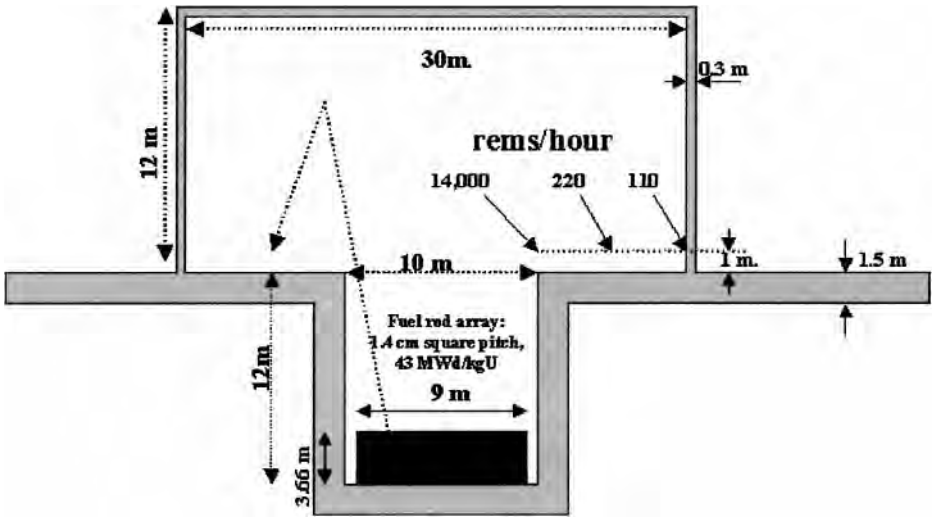


**Figure 6:** Calculated radiation levels from a drained spent-fuel pool one meter above the level of the floor of a simplified cylindrically-symmetric spent-fuel-pool building. Even out of direct sight of the spent fuel, the radiation dose rates from gamma rays scattered by the air, roof and walls are over a hundred rems/hr.

wings into runways. They concluded that at speeds above 60 m/s (135 mph), approximately

> 50% of the liquid is so finely atomized that it evaporates before reaching the ground. If this were fuel, a fireball would certainly have been the result, and in the high-temperature environment of the fireball a substantially larger fraction of the mass would have evaporated.[39]

The blast that would result from such a fuel-air explosion might not destroy the pool but could easily collapse the building above, making access difficult and dropping debris into the pool. A potentially destructive fuel-air deflagration could also occur in spaces below some pools. Any remaining kerosene would be expected to pool and burn at a rate of about 0.6 cm/minute if there is a good air supply.[40]

The burning of 30 cubic meters of kerosene—about one third as much as can be carried by the type of aircraft which struck the World Trade Center on September 11, 2001[41]—would release about $10^{12}$ joules of heat—enough to evaporate 500 tons of water. However, under most circumstances, only a relatively small fraction of the heat would go into the pool.

*Puncture by an Airplane Engine Turbine Shaft, Dropped Cask or Shaped Charge*

As Figure 2 suggests, many spent-fuel pools are located above ground level or above empty cavities. Such pools could drain completely if their bottoms were punctured or partially if their sides were punctured.

Concerns that the turbine shaft of a crashing high-speed fighter jet or an act of war might penetrate the wall of a spent-fuel storage pool and cause a loss of coolant led Germany in the 1970s to require that such pools be sited with their associated reactors inside thick-walled containment buildings. When Germany decided to establish large away-from-reactor spent-fuel storage facilities, it rejected large spent-fuel storage pools and decided instead on dry storage in thick-walled cast-iron casks cooled on the outside by convectively circulating air. The casks are stored inside reinforced-concrete buildings that provide some protection from missiles.[42]

Today, the turbine shafts of larger, slower-moving passenger and freight aircraft are also of concern. After the September 11, 2001 attacks against the World Trade Center, the Swiss nuclear regulatory authority stated that

> From the construction engineering aspect, nuclear power plants (worldwide) are *not* protected against the effects of warlike acts or terrorist attacks from the air. . . . one cannot rule out the possibility that fuel elements in the fuel pool or the primary cooling system would be damaged and this would result in a release of radioactive substances [emphasis in original][43]

The NRC staff has decided that it is prudent to assume that a turbine shaft of a large aircraft engine could penetrate and drain a spent-fuel-storage pool.[44] Based on calculations using phenomenological formulae derived from experiments with projectiles incident on reinforced concrete, penetration cannot be ruled out for a high-speed crash but seems unlikely for a low-speed crash.[45]

This is consistent with the results of a highly-constrained analysis recently publicized by the Nuclear Energy Institute (NEI).[46] The analysis itself has not been made available for independent peer review "because of security considerations." According to the NEI press release, however, it concluded that the engine of an aircraft traveling at the low speed of the aircraft that struck the Pentagon on Sept. 11, 2001 (approximately 350 miles/hr or 156 m/s) would not penetrate the wall of a spent-fuel-storage pool. Crashes at higher speed such as that against the World Trade Center South Tower (590 miles/hr or 260 m/s), which had about three times greater kinetic energy, were ruled out because the "probability of the aircraft striking a specific point on a structure—particularly one of the small size of a nuclear plant—is significantly less as speed increases."

The NEI press release included an illustration showing a huge World Trade Center tower (63 meters wide and 400 meters tall) in the foreground and a tiny spent-fuel pool (24 meters wide and 12 meters high) in the distance. Apparently no analysis was undertaken as to the possibility of a crash destroying the supports under or overturning a spent-fuel pool. A less constrained analysis should be carried out under U.S. Government auspices.

A terrorist attack with a shaped-charge anti-tank missile could also puncture a pool—as could a dropped spent-fuel cask.[47]

## COOLING PROCESSES IN A PARTIALLY OR FULLY-DRAINED SPENT-FUEL POOL

### "Dense packing"

U.S. storage pools—like those in Europe and Japan—were originally sized on the assumption that the spent fuel would be stored on site for only a few years until it was cool enough to transport to a reprocessing plant where the fuel would be dissolved and plutonium and uranium recovered for recycle. In 1974, however, India tested a nuclear explosive made with plutonium recovered for "peaceful" purposes. The Carter Administration responded in 1977 by halting the licensing of an almost completed U.S. reprocessing plant. The rationale was that U.S. reprocessing might legitimize the acquisition of separated plutonium by additional countries interested in developing a nuclear-weapons option. In the 1982 Nuclear Waste Policy Act, therefore, the U.S. Government committed to provide an alternative destination for the spent fuel accumulating in reactor pools by building a deep-underground repository. According to the Act, acceptance of spent fuel at such a repository was supposed to begin by 1998. As of this writing, the US Department of Energy (DoE) projects that it can open the Yucca Mountain repository in 2010[48] but the US General Accounting Office has identified several factors, including budget limitations, that could delay the opening to 2015 or later.[49]

U.S. nuclear-power plant operators have dealt with the lack of an off-site destination for their accumulating spent fuel by packing as many fuel assemblies as possible into their storage pools and then, when the pools are full, acquiring dry storage casks for the excess. The original design density of spent fuel in the pools associated with PWRs had the fuel assemblies spaced out in a loose square array. The standard spacing for new dense-pack racks today is 23 cm—barely above the 21.4 cm spacing in reactor cores.[50] This "dense-packed" fuel is kept sub-critical by enclosing each fuel assembly in a metal box whose walls contain neutron-absorbing boron[51] (see Figure 7[52]).

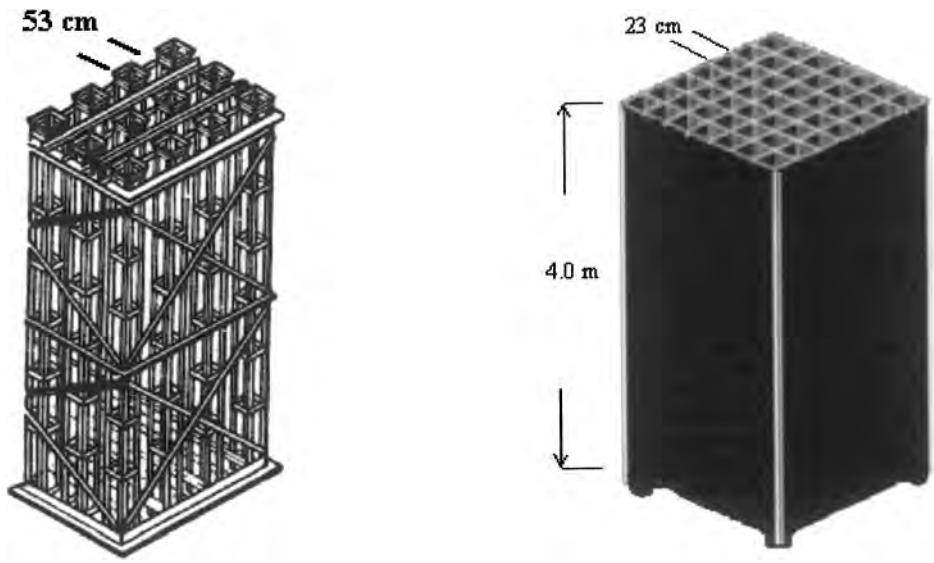


**Figure 7:** Open and dense-pack PWR spent-fuel racks (*Sources:* Left: NUREG/CR-0649, SAND77-1371, 1979; right: authors).

These boron-containing partitions would block the horizontal circulation of cooling air if the pool water were lost, greatly reducing the benefits of mixing recently-discharged with older, cooler fuel. During a partial uncovering of the fuel, the openings at the bottoms of the spent-fuel racks would be covered in water, completely blocking air from circulating up through the fuel assemblies. The portions above the water would be cooled primarily by steam produced by the decay heat in the below-surface portions of the fuel rods in the assemblies and by blackbody radiation.[53]

In the absence of *any* cooling, a freshly-discharged core generating decay heat at a rate of 100 kWt/tU would heat up adiabatically within an hour to about 600°C, where the zircaloy cladding would be expected to rupture under the internal pressure from helium and fission product gases,[54] and then to about 900°C where the cladding would begin to burn in air.[55] It will be seen that the cooling mechanisms in a drained dense-packed spent-fuel pool would be so feeble that they would only slightly reduce the heatup rate of such hot fuel.

In 2001, the NRC staff summarized the conclusions of its most recent analysis of the potential consequences of a loss-of-coolant accident in a spent fuel pool as follows:

> [I]t was not feasible, without numerous constraints, to establish a generic decay heat level (and therefore a decay time) beyond which a zirconium fire is

> physically impossible. Heat removal is very sensitive to . . . factors such as fuel assembly geometry and SFP [spent fuel pool] rack configuration . . . [which] are plant specific and . . . subject to unpredictable changes after an earthquake or cask drop that drains the pool. Therefore, since a non-negligible decay heat source lasts many years and since configurations ensuring sufficient air flow for cooling cannot be assured, the possibility of reaching the zirconium ignition temperature cannot be precluded on a generic basis.[56]

We have done a series of "back-of-the-envelope" calculations to try to understand the computer-model calculations on which this conclusion is based. We have considered thermal conduction, infrared radiation, steam cooling, and convective air cooling.

*Thermal Conduction*

Conduction through the length of uncovered fuel could not keep it below failure temperature until the fuel had cooled for decades.[57]

*Infrared Radiation*

Infrared radiation would bring the exposed tops of the fuel assemblies into thermal equilibrium at a temperature of $T_0 = [PM/(A\sigma)]^{1/4}$ °K, where P is the power (Watts) of decay heat generated per metric ton of uranium, M is the weight of the uranium in the fuel assembly (0.47 tons), $A = 500$ cm$^2$ is the cross-sectional area of the dense-pack box containing the fuel assembly, and $\sigma$ ($= 5.67 \times 10^{-12}$ $T_K^4$ Watts/cm$^2$) is the Stefan-Boltzman constant. (We assume that the top of the fuel assembly radiates as a black body, i.e., maximally.) For $P = 1$ kW or 10 kW, $T_0$ is respectively 370 or 860°C.

With radiative cooling only, however, the temperatures in the depths of the fuel assemblies would be much hotter, because most of the radiation from the interior of the fuel would be reabsorbed and reradiated by other fuel rods many times before it reached the top end of the fuel assembly. Even for $P = 1$ kW/tU (roughly 30-year-old fuel) the temperature at the bottom of the fuel assembly would be about 2000°C.[58] Therefore, while radiation would be effective in cooling the exposed surfaces of older fuel assemblies, it would not be effective in cooling their interiors.

*Steam Cooling*

Steam cooling could be effective as long as the water level covers more than about the bottom quarter of the spent fuel. Below that level, the rate of steam generation by the fuel will depend increasingly on the rate of heat transfer

from the spent fuel to the water via blackbody radiation. The rate at which heat is transferred directly to the water will decline as the water level sinks and the temperature of the fuel above will climb. When the water is at the bottom of the fuel assembly, it appears doubtful that this mechanism could keep the peak temperature below 1200°C for fuel less than a hundred years post discharge.[59] Since even steels designed for high-temperature strength lose virtually all their strength by 1000°C and zircaloy loses its strength by 1200°C, the tops of the racks could be expected to begin to slump by the time this water level is reached.[60]

*Convective Air Cooling*

After a complete loss of coolant, when air could gain access to the bottom of the fuel assemblies, convective air cooling would depend upon the velocity of the air through the fuel assemblies. The heat capacity of air is about 1000 joules/kg-°C, its sea-level density at a 100°C (373°K) entrance temperature into the bottom of a fuel assembly is about 0.9 kg/m$^3$, the cross-section of the portion of a dense-pack box that is not obstructed by fuel rods would be about 0.032 m$^2$,[61] and each fuel assembly contains about 0.47 tons of uranium. The vertical flow velocity of air at the bottom of the assembly for an air temperature rise to 900°C (1173°K) then would be 0.023 m/sec per kW/tU. Because the density of the air varies inversely with its absolute temperature, this velocity would increase by a factor of $(1173/373) \approx 3$ at the top of the fuel assembly.

The pressure accelerating the air to this velocity would come from the imbalance in density—and therefore weight—of the cool air in the space between the fuel racks and the pool wall (the "down-comer") and the warming air in the fuel assemblies. If we assume that the density of the air in the down-comer is 1 kg/m$^3$ and that it has an average density of 0.5 kg/m$^3$ in the fuel assemblies, then the weight difference creates a driving pressure difference. Neglecting friction losses, this pressure difference would produce a velocity for the air entering the bottom of the fuel assembly of about 2.7 m/s, sufficient to remove heat at a rate of 120 kW/tU. Adding friction losses limits the air velocity to about 0.34 m/s, however, which could not keep PWR fuel below a temperature of 900°C for a decay heat level greater than about 15 kW/tU—corresponding to about a year's cooling.[62] Adding in conductive and radiative cooling would not change this result significantly.

This is consistent with results obtained by more exact numerical calculations that take into account friction losses in the down-comer and the heating of the air in the building above the spent-fuel pool.[63] The 1979 Sandia study obtained similar results. It also found that, in contrast to the situation with

dense-pack storage, with open-frame storage and a spacing between fuel assemblies of 53 cm (i.e., a density approximately one fifth that of dense-packed fuels), convective air cooling in a well-ventilated spent-fuel storage building (see below) could maintain spent fuel placed into the spent-fuel pool safely below its cladding failure temperature as soon as 5 days after reactor shutdown.[64] These important conclusions should be confirmed experimentally with, for example, electrically heated fuel rods.[65]

*Spread of Fires from Hot to Colder Fuel*

The above discussion has focused on the likelihood that recently-discharged dense-packed fuel could heat up to ignition temperature in either a partially or fully drained pool. It is more difficult to discuss quantitatively the spread of such a fire to adjacent cells holding cooler fuel that would not ignite on its own. A 1987 Brookhaven report attempted to model the phenomena involved and concluded that "under some conditions, propagation is predicted to occur for spent fuel that has been stored as long as 2 years."[66] The conditions giving this result were dense-packing with 5 inch [13 cm] diameter orifices at the bottom of the cells—i.e., typical current U.S. storage arrangements.

The report notes, however, that its model

> does not address the question of Zircaloy oxidation propagation after clad melting and relocation [when] a large fraction of the fuel rods would be expected to fall to the bottom of the pool, the debris bed will remain hot and will tend to heat adjacent assemblies from below [which] appears to be an additional mechanism for oxidation propagation.

The report therefore concludes that the consequences of two limiting cases should be considered in estimating the consequences of spent-fuel pool fires: 1) only recently discharged fuel burns, and 2) all the fuel in the pool burns.[67] This is what we have done above. We would add, however, that any blockage of air flow in the cooler channels of a dense-packed pool by debris, residual water, or sagging of the box structure would facilitate the propagation of a spent-fuel fire.[68]

## MAKING SPENT-FUEL POOLS, THEIR OPERATION, AND THEIR REGULATION SAFER

A variety of possibilities can be identified for reducing the risk posed by spent-fuel pools. Some were considered in reports prepared for the NRC prior to the

Sept. 11, 2001 destruction of the World Trade Center and rejected because the estimated probability of an accidental loss of coolant was so low (about 2 chances in a million per reactor year) that protecting against it was not seen to be cost effective.[69]

Now it is necessary to take into account the potentially higher probability that a terrorist attack could cause a loss of coolant. Since the probabilities of specific acts of malevolence cannot be estimated in advance, the NRC and Congress will have to make a judgment of the probability that should be used in cost-benefit analyses. The most costly measures we propose would be justified using the NRC's cost-benefit approach if the probability of an accident or attack on a U.S. spent-fuel pool resulting in a complete release of its $^{137}$Cs inventory to the atmosphere were judged to be 0.7 percent in a 30-year period. *This is at the upper end of the range of probabilities estimated by the NRC staff for spent-fuel fires caused by accidents alone.* For a release of one tenth of the $^{137}$Cs inventory, the break-even probability would rise to about 5 percent in 30 years.[70]

Below, we discuss more specifically initiatives to:

- Reduce the probability of an accidental loss of coolant from a spent-fuel pool,
- Make the pools more resistant to attack,
- Provide emergency cooling,
- Reduce the likelihood of fire should a loss of coolant occur, and
- Reduce the inventory of spent fuel in the pools.

Included are three recommendations made in the 1979 Sandia study on the consequences of possible loss-of-coolant accidents at spent-fuel storage pools.[71] Unfortunately, all of these approaches offer only partial solutions to the problem of spent-fuel-pool safety. That problem will remain as long as nuclear power plants operate. However, the probability of a spent-fuel fire can be significantly reduced, as can its worst-case consequences. Some options will involve risk tradeoffs, and will therefore require further analysis before decisions are made on their implementation.

We discuss the specific changes below under three headings: regulatory, operational, and design.

## Regulatory

NRC regulations do not currently require either qualified or redundant safety systems at spent-fuel pools or emergency water makeup capabilities.[72] The

NRC should require reactor owners to remedy this situation and demonstrate the capability to operate and repair spent-fuel pools and their supporting equipment under accident conditions or after an attack. This capability would contribute to defense in depth for nuclear power plants and spent fuel.[73]

## Operational

*Minimize the Movement of Spent-Fuel Casks Over Spent-Fuel Pools*

The NRC staff study, *Spent Fuel Accident Risk*, concludes that "spent fuel casks are heavy enough to catastrophically damage the pool if dropped." The study cites industry estimates that casks are typically moved "near or over the SFP (spent fuel pool) for between 5 and 25 percent of the total path." It was concluded that this was not a serious concern, however, because industry compliance with NRC guidance would result in the probability of a drop being reduced to less than $10^{-5}$ per reactor-year.[74] Nevertheless, we recommend consideration of whether the movements of spent-fuel casks over pools can be reduced. We also acknowledge that reducing a pool's inventory of fuel, as recommended below, will increase the number of cask movements in the near term—although all the fuel will eventually have to be removed from the pools in any case. The resulting risk increase should be minimized as part of the implementation plan.

*Minimize Occasions When the Entire Core is Moved to the Pool During Refueling Outages*

Refueling outages occur every 12 to 18 months and typically last a month or so. Pool dry-out times decrease dramatically when full cores are placed into spent-fuel-storage pools only a few days after reactor shutdown. Only a third to a quarter of the fuel in the core is actually "spent." The remainder is moved back into the core at new positions appropriate for its reduced fissile content. It is not necessary to remove the entire core to the spent fuel pool to replace the fuel assemblies in their new locations.[75] Even when it is necessary to inspect the interior of the pressure vessel or to test the fuel for leakage, removal of part of the fuel should be adequate in most cases. The only regulatory *requirement* for removal of the entire core is on those infrequent occasions when work is being done that has the potential for draining the reactor pressure vessel. This would be the case, for example, when work is being done on a pipe between the

pressure vessel and the first isolation valve on that pipe—or on the isolation valve itself.[76]

## Design

*Go to Open-Frame Storage*

As already noted, the Sandia study found that, for pools with open-frame storage in well-ventilated storage buildings (see below), spent fuel in a drained storage pool will not overheat if it is cooled at least 5 days before being transferred to the pool. Furthermore, for partial drainage, which blocks air flow from below, open-frame storage allows convective cooling of the fuel assemblies from the sides above the water surface.

The simplest way to make room for open-frame storage at existing reactors is to transfer all spent fuel from wet to dry storage within five years of discharge from the reactor. Consequently, our proposal for open-frame storage is tied to proposals for dry storage, as discussed below.

The open-frame storage considered in the Sandia study could store, however, only 20 percent as much fuel as a modern dense-pack configuration. Thus, a pool that could hold 500 tons of dense-packed spent fuel from a 1000-MWe unit could accommodate in open racks the approximately 100 tons of spent fuel that would be discharged in five years from that reactor.[77] However, about twice as large a pool would be required to provide enough space in addition to accommodate the full reactor core in open-frame storage. If this much space were not available, occasions in which a full-core discharge is required would remain dangerous—although less frequent, if the recommendation to minimize full-core offloads is adopted.

Alternative approaches to a lack of sufficient space for open-rack storage would be to move spent fuel out of the pool earlier than five years after discharge or to adopt racking densities intermediate between dense-pack and the Sandia open rack arrangement. Two interesting intermediate densities that should be analyzed are: 1) an arrangement where one fifth of the fuel assemblies are removed in a pattern in which each of the remaining fuel assemblies has one side next to an empty space; 2) an arrangement where alternate rows of fuel assemblies are removed from the rack. These geometries would have to include perforations in the walls to allow air circulation in situations where enough water remained in the pool to block the openings at the bottoms of the boxes, or removal of some partitions entirely.

One problem with open-rack storage is that it creates a potential for a criticality accident for fresh or partially burned fuel if the fuel racks are crushed.

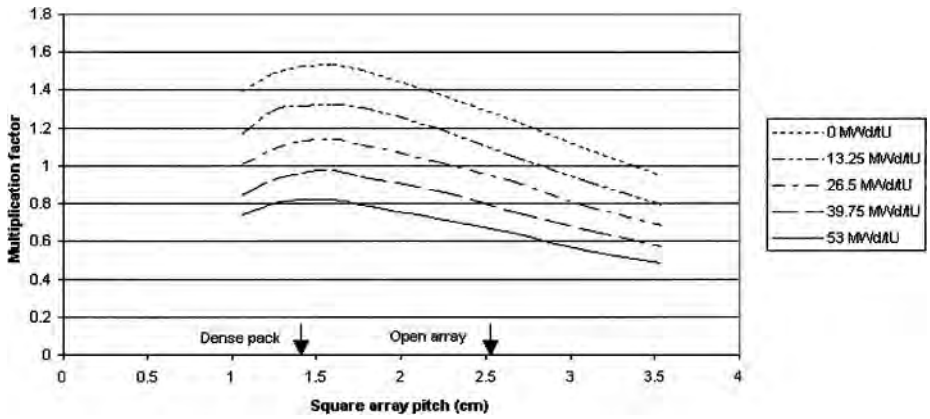


**Figure 8:** Neutron multiplication as a function of array pitch in an infinite square array of 4.4% enriched fuel rods with a design burnup of 53 MWd/kgU for 0, 25, 50, 75 and 100% irradiation (*Source:* authors).

Figure 8 shows the value of the neutron multiplication factor $k_{eff}$ in an infinite square array of 4.4% enriched fuel at various burnups as a function of the spacing between the rod centers (the array "pitch") in a pool of unborated water.[78] It will be seen that, for burnups of less than 50 percent, the open array is critical at a pitch of 2.6 cm and that the neutron multiplication factor increases as the pitch decreases to about 1.6 cm.

This situation is most problematical for low-burnup fuel. One way to remedy the situation for low-burnup fuel would be to put in neutron-absorbing plates between rows of fuel assemblies.[79] This would still allow free convection of air through the rows. Other configurations of neutron-absorbing material could also be consistent with allowing free convection. Suppression of criticality could also be achieved by adding a soluble compound of neutron-absorbing boron to the pool water.[80] Finally, some high-density rack spaces could be provided for low-burnup fuel. If fresh fuel is stored in pools, it could certainly be put in dense-rack storage since fresh fuel does not generate significant heat.

*Provide for Emergency Ventilation of Spent-Fuel Buildings*

The standard forced air exchange rate for a spent-fuel-storage building is two air changes per hour.[81] Consider a building with an air volume $V$ and an air exchange rate of $n$ volumes of external air per hour. If the spent fuel generates heat at a rate P, the air temperature rise will be $\Delta T = 3600P/(nV\rho c_p)$ where $\rho$ is the density of the air entering the building (about 1 kg/m$^3$) and $c_p$ is the

heat capacity of the air per kg at constant pressure [(about 1000 joules/(kg-°C)]. Therefore, $\Delta T \sim 3.6P/(nV)$. Consider a case where the spent-fuel pool contains 80 tons of freshly-discharged fuel generating 100 kWt/tU of decay heat (i.e., P = 8 MWt) and where V = 10,000 cubic meters (e.g., a building roughly 30 meters square and 10 meters tall). For this case, $\Delta T \sim 2900/n°C$. To bring $\Delta T$ down to 100°C would require about 30 air exchanges per hour.

The Sandia report proposed that, in case of a loss-of-coolant accident, large vents in the sides and roof of the building be opened to allow a high rate of convective air exchange. The required area of the openings was calculated by equating the outside-inside air pressure difference at the floor of a building H meters high due to the difference in air densities outside and inside: $\Delta p = gH(\rho_o - \rho_i)$ with the sum of the throttling pressure losses at the openings: $\Delta p_{th} = 0.5\rho_o(v_i/C_D)^2 + 0.5\rho_i(v_o/C_D)^2$. Here $v_i$ and $v_o$ are respectively the average velocities of the incoming and exiting air and the "discharge coefficient," $C_D \sim 0.6$, reflects the reduction of the air velocity due to turbulence caused by the edges of the openings. Taking into account the fact that air density varies inversely with absolute temperature, the minimum area of the openings can be calculated as[82]

$$A = \{P/[C_D c_p \rho_o (2gH)^{1/2}]\}\{T_i(T_o + T_i)/[T_o(\Delta T)^3]\}^{1/2}$$

For H = 10 m, $T_i = 300°K$ and $\Delta T = 100°K$, this equation becomes $A = 3.6P\ m^2$ if P is measured in megawatts. Thus, if P = 8 MWt, A would have to be 30 $m^2$, e.g. an opening 10 meters long and 3 meters high.

Of course, such a system would not prevent a fire in a dense-packed pool because of the poor air circulation in the spent-fuel racks. It is a complement to open-rack storage, not a substitute.

The venting system design proposed in the Sandia report is attractive because it is passive. However, it might be difficult to retrofit into existing buildings, the door-opening system might be incapacitated, and it would not work if the building collapsed as a result of an accident or terrorist act. Furthermore, if a fire did start, the availability of ventilation air could feed the fire. Therefore, high-capacity diesel-powered blowers should be considered as an alternative or complement to a passive ventilation system.

*Install Emergency Water Sprays*

The Sandia report also proposed that a sprinkler system be installed.[83] For 80 tons of spent fuel generating 100 kWt/MTU, the amount of water required if it were all evaporated would be about 3 liters per second. Such a flow could easily

be managed in a sprinkler system with modest-sized pipes.[84] The sprinkler system should be designed with an assured supply of water and to be robust and protected from falling debris. It should also be remotely operated, since the radiation level from uncovered fuel would make access to and work in a spent-fuel building difficult to impossible—especially if the building were damaged. The hottest fuel should be stored in areas where spray would be the heaviest, even if the building collapses on top of the pool (e.g., along the sides of the pool). The spray would need to reach all of the spent fuel in the pool, however—especially in scenarios where the spray water accumulated at the bottom of the pool and blocked air flow into the dense-pack racks.

Another circumstance in which the spray could aggravate the situation would be if the spent-fuel racks were crushed or covered with debris, blocking the flow of air. In such a case, steam generated from water dripping into the superheated fuel could react with the zirconium instead. The circumstances under which sprays should be used would require detailed scenario analysis.

*Make Preparations for Emergency Repairs of Holes*

A small hole, such as might be caused by the penetration of a turbine shaft or an armor-piercing warhead, might be patched. For a hole in the side, a flexible sheet might be dropped down the inside of the pool.[85] However, in the turbine-shaft case, the space might be blocked if the projectile was protruding from the wall into the spent-fuel rack. Or the racks might be damaged enough to close the gap between them and the side of the pool. Also, if the top of the fuel were already exposed, the radiation levels in the pool area would be too high for anything other than pre-emplaced, remotely controlled operations.

Patching from the outside would be working against the pressure of the water remaining in the pool (0.1 atmosphere or 1 kg/cm$^2$ per meter of depth above the hole). However, there could be better access and the pool wall would provide shielding—especially if the hole were small. Techniques that have been developed to seal holes in underground tunnels might be useful.[86]

*Armor Exposed Outside Walls and Bottoms Against Projectiles*

The water and fuel in the pool provide an effective shield against penetration of the pool wall and floor from the inside. It should be possible to prevent penetration by shaped charges from the outside with a stand-off wall about 3 meters away that would cause the jet of liquid metal formed by the shaped charge to expand and become much less penetrating before it struck the pool wall. In the case of the turbine shaft, Pennington's analysis for dry casks suggests that it

also might be possible to absorb the shaft's energy with a thick sheet of steel that is supported in a way that allows it to stretch elastically and absorb the projectile's kinetic energy (see below).

## REDUCING THE INVENTORY OF SPENT-FUEL POOLS

Our central proposal is to move spent fuel into dry storage casks after it has cooled for 5 years.[87] In addition to allowing for a return to open-frame storage, such a transfer would reduce the typical $^{137}$Cs inventory in a pool by approximately a factor of four,[88] thereby reducing the worst-case release from a pool by a comparable factor. Casks are already a growing part of at-reactor storage capacity. Out of the 103 operating power reactors in the U.S., 33 already have dry cask storage and 21 are in the process of obtaining dry storage.[89] On average about 35 casks would be needed to hold the 5-year or more aged spent fuel in a spent fuel pool filled to capacity.[90]

As already noted, to a certain extent this proposal runs counter to the earlier proposal to mimimize the movement of spent fuel casks over pools. The risk of dropped casks should be considered in deciding on which types of dry storage transfer casks are utilized.

## SAFETY OF DRY-CASK STORAGE

Shifting pools back toward open-rack storage would require moving much of the spent fuel currently in pools into dry storage casks. With currently licensed casks, this could be done by the time the fuel has cooled 5 years.

In principle, the transfer of the spent fuel to dry storage could take place earlier. Spent fuel cooled for 2.5 years has about twice the decay heat per ton as spent fuel 5 years after discharge (see figure 5). Such spent fuel might be stored next to the walls of storage casks with older, cooler spent fuel stored in the interior.

Casks are not vulnerable to loss of coolant because they are cooled by natural convection that is driven by the decay heat of the spent fuel itself. Thus dry-storage casks differ from reactors and existing spent-fuel pools in that their cooling is completely passive. To obtain a release of radioactive material, the wall of the fuel container must be penetrated from the outside, or the container must be heated by an external fire to such an extent that the containment envelope fails. However, many dry-storage modules must fail or be attacked simultaneously to produce the very large releases that are possible today at spent-fuel pools. Nevertheless, since the total $^{137}$Cs inventory on-site does not

change under our proposal, it is important to examine the safety of dry-cask storage as we envisage it being used.

There are two basic types of dry storage cask currently licensed in the U.S. (see Figure 9):[91]

1. Casks whose walls are thick enough to provide radiation protection; and
2. Thin-walled canisters designed to be slid into a concrete storage overpack that provides the radiation shielding with space between the cask and overpack for convective circulation of air. (Transfer overpacks and transport overpacks are used for onsite movement and offsite shipping, respectively.)

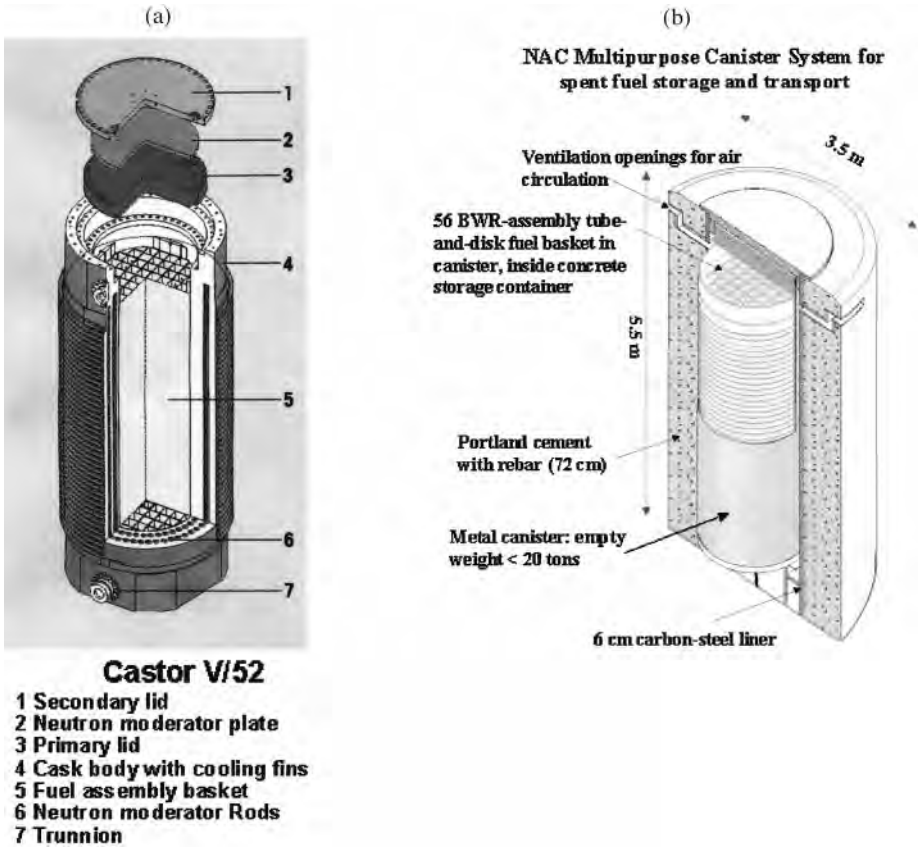


**Figure 9:** (a) Thick-walled cask[103] and (b) Cask with overpack.[104] (*Sources:* GNB and NAC).

Among the possible threats to such casks are: shaped-charge missiles, aircraft turbine spindles, and fire.

## Shaped-Charge Missile

Dry storage casks in the U.S. are stored on concrete pads in the open. Missiles tipped with shaped charges designed to penetrate tank armor could penetrate such an unprotected storage cask and cause some damage to the fuel within. Experiments on CASTOR-type spent fuel casks of 1/3 length and containing a $3 \times 3$ array of assemblies were carried out in 1992 at a French army test site for Germany's Ministry of the Environment and Nuclear Safety (BMU). The simulated fuel was made of unirradiated depleted uranium pressurized to 40 atmospheres to simulate the pressure buildup from fission product gases in spent fuel.

The particulate matter released through the hole was collected and analyzed for size distribution. When the initial pressure within the cask was atmospheric, about 3.6 grams of particles with diameters less than 100 microns were released in a puff from the hole. In the analysis of radiological consequences, it was assumed that, because of its volatility, $^{137}$Cs equivalent to that in 50 grams of spent fuel with a burnup of 48.5 MWd/tU would be released.[92] Another analysis assumed a $^{137}$Cs release 1000 times larger.[93] A still larger release could occur if a cask were attacked in such a way as to initiate and sustain combustion of the zirconium cladding of the fuel.

It has been found possible to plug the relatively small hole made by a shaped charge in a thick-walled iron cask with a piece of lead before much radioactivity could be released.[94] Plugging the hole would be considerably more difficult in the case of a thin-walled cask surrounded by a concrete overpack.

In each case, unless the fuel-in a significant fraction of the casks were ignited, the release would be small in comparison to the potential release resulting from a spent-fuel-pool fire. Nevertheless, German authorities require casks to be stored inside a shielding building. The building walls could be penetrated by a shaped charge but the liquid metal would spread in the space between the wall and the nearest cask and therefore be relatively harmless. U.S. dry-cask storage areas are not currently so protected but the casks could be protected with an overpack[95] and/or a berm.

## Turbine Spindle

The Castor cask has survived, without penetration impacts, from various angles by a simulated turbine spindle weighing about half a ton surrounded by additional steel weighing about as much and traveling at almost sonic speed

(312 m/sec).[96] Recently, NAC International carried out a computer simulation of the impact of a Boeing 747 turbine on its canister-in-overpack Universal Multipurpose System at a speed of 220 m/sec and concluded it too would not be penetrated. This conclusion should be verified experimentally.[97]

### Fire

Theoretical studies of the resistance to fire of Castor V/19 (PWR) and V/52 (BWR) storage/transport casks were done for Austria's Environmental Agency for a number of German reactor sites because of concerns that the contamination from cask failure might extend into Austria. The scenario was a crash of a large commercial airliner into a storage facility. It was assumed that 60 tons of kerosene pooled around the storage casks and burned for 3 to 5 hours at a temperature of 1000°C. It was estimated that, because of the massive heat capacity of the thick cask walls, the seals of their bolted-down lids would begin to fail only after 3 hours. It was also assumed that, by that time, the fuel cladding would have failed. Finally, it was assumed that the contained $^{137}$Cs would be in its most volatile possible (elemental) form. On this basis, it was estimated that about 0.04 MCi of $^{137}$Cs would be released after a 5-hour, 1000°C fire in a storage facility with 135 casks containing a total of 170 MCi.[98]

Obviously, the release from even such a worst-case incident would be tiny compared with the 100 to 1000 times higher releases from a spent-fuel pool fire considered above. However, a spent-fuel storage facility should be designed, among other requirements, to prevent the pooling of kerosene around the casks.

## IMPLEMENTATION ISSUES RELATING TO THE TRANSFER OF OLDER SPENT FUEL TO DRY-CASK STORAGE

As will be explained, given existing cask-production capacity, it would take about a decade to move most of the spent fuel currently in pools into dry-cask storage. Virtually all of the storage would have to be at the reactor sites for some decades until off-site disposal becomes available. The Yucca Mountain underground repository will not open for at least a decade and current plans have spent fuel being shipped to the repository at a rate of 3000 tons per year—only about 1000 tons/yr more than the current rate of spent-fuel discharge from U.S. reactors.[99] If the opening of Yucca Mountain is delayed for many years, approximately 2000 tons of spent fuel per year might be shipped to a proposed large centralized facility on the Goshute reservation west of Salt Lake City, Utah—if it is licensed.[100]

For comparison, the inventory of spent fuel at U.S. reactor sites will be more than 60,000 tons in 2010, of which about 45,000 tons will be in mostly dense-packed pools.[101] If all but the last 5 years of discharges are dry stored, approximately 35,000 tons will have to be unloaded from the pools.[102] Since it would be imprudent to assume that off-site shipments to Yucca Mountain or a centralized interim spent-fuel storage facility could be relied on to solve the problem of dense-packed spent-fuel pools anytime soon, we focus here on the logistical and cost issues associated with increasing the amount of on-site dry storage.

## Cask Availability

Cask availability could be a rate-limiting step in moving older spent fuel from pools into dry storage at the reactor sites. Currently, US cask fabrication capacity is approximately 200 casks per year—although the production rate is about half that. Two hundred casks would have a capacity about equal to the spent-fuel output of U.S. nuclear power plants of about 2000 tons per year. However, according to two major U.S. manufacturers, they could increase their combined production capacity within a few years to about 500 casks per year.[105] To use the extra 300 casks per year to unload 35,000 tons of spent fuel out of the storage pools would require about 10 years. This period could be reduced somewhat if the unloading of high-density pools was perceived to be an important issue of homeland security. The United States has substantial industrial capacity that could be allocated to cask production using existing, licensed designs. Casks made in Europe and Japan could be imported as well. However, other potentially rate-limiting factors would also have to be considered in any estimate of how much the transfer period could be shortened.

## Dry-Storage Costs

Storage cask capacity costs U.S. utilities from $90 to $210/kgU.[106] Additional capital investments for new on-site dry storage facilities would include NRC licensing, storage pads, security systems, cask welding systems, transfer casks, slings, tractor-trailers, and startup testing. These costs are estimated to range from $9 to $18 million per site.[107] However, at most sites, they will be incurred in any case, since even dense-packed pools are filling up. The capital cost of moving 35,000 tons of spent fuel into dry casks would therefore be dominated by the cost of the casks and would range from about $3.5 to $7 billion ($100–200/kgU). Per GWe of nuclear capacity, the cost would be $35–70 million. The additional cost per kWh would be about 0.03–0.06 cents/kWh.[108] This is 0.4–0.8 percent of the average retail price of electricity in 2001.[109] It is also

equivalent to 30 to 60 percent of the federal charge for the ultimate disposition of the spent fuel (see below).

The extra cost would be reduced significantly if the casks could be used for transport and ultimate disposal as well. For multi-purpose canisters with stationary concrete overpacks, the extra cost would then be associated primarily with the overpack (about 20% of the total cost) and with the need to buy the canisters earlier than would have been the case had the spent fuel stayed in dense-packed pools until it was transported to the geological repository. Unfortunately, the Department of Energy has abandoned the idea of multi-purpose containers and currently plans to have spent fuel unpacked from transport canisters and then repacked in special canisters for disposal.[110]

Costs would be increased by the construction of buildings, berms or other structures to surround the casks and provide additional buffering against possible attack by anti-tank missiles or crashing aircraft. The building at Gorleben, which is licensed to hold 420 casks containing about 4200 tons of uranium in spent fuel, would cost an estimated $20–25 million to build in the United States or about $6/kgU.[111] Assuming conservatively that the building cost scales with the square root of the capacity (i.e. according to the length of its walls), it would cost about $12/kgU for a facility designed to store 100 casks containing 1000 tons uranium in spent fuel—about the inventory of a typical 2-reactor site if our proposal was carried through by 2010.[112] Berms for a middle-sized storage area might cost about $1.5–3/kgU.[113]

## Licensing Issues

The NRC currently licenses storage casks for 20 years. Some U.S. dry-cask storage facilities will reach the 20-year mark in a few years. The NRC is therefore currently deciding what analysis will be required to provide a basis for license extensions.

With reactor operators increasing fuel burnup, casks will also eventually have to be licensed for the storage of high-burnup fuel. Current licenses allow burnups of up to 45,000 MWd/MT. However, the CASTOR V/19 cask is already licensed in Germany to store 19 high-burnup Biblis-type fuel assemblies, which are slightly bigger and heavier than U.S. PWR fuel assemblies. The license allows 15 five-year cooled fuel assemblies with burnups of 55 MWd/kgU plus four with burnups of up to 65 MWd/kgU.[114] U.S. storage casks have been tested with fuels with burnups of 60 MWd/kgU.[115]

Finally, some reactor operators have expressed concern that the NRC does not currently have sufficient manpower to accelerate the process of licensing

on-site dry storage. However, almost all sites will have to license dry storage in the timeframe considered here in any case.

## Who Will Pay?

Nuclear power operators can be expected to balk at the extra cost of moving spent fuel out of pools to on-site dry storage. As a result of deregulation, many operators are no longer able to pass such costs through to customers without fear of being undersold by competing fossil-fueled power plants. Also, many plants have been sold at a few percent of their original construction costs to owners who have established corporations to limit their liability to the value of the plants themselves.[116] Therefore, to prevent extended delays in implementing dry storage, the federal government should consider offering to pay for extra storage casks and any security upgrades that it might require for existing dry storage facilities.

Under the Nuclear Waste Policy Act (NWPA) of 1982, the Department of Energy (DoE) was to enter into contracts with nuclear utilities to begin moving spent fuel from nuclear power plants to a national deep underground repository by 1998. In exchange, the utilities made payments to a national Nuclear Waste Fund at the rate of 0.1 cents per net electrical kilowatt-hour generated by their nuclear plants plus a one-time payment (which some utilities have not yet fully paid) based on their nuclear generation prior to the law's enactment. As of May 31, 2002, this fund had a balance of $11.9 billion. Since 1995, $600–700 million have been deposited annually.[117] The DoE spends about $600 million annually on Yucca Mountain but, for the past several years, about two thirds of this amount has been drawn from the National Defense Account of the U.S. Treasury because the DoE had previously underpaid for the share of the facility that will be occupied by high-level radioactive waste from its defense nuclear programs.

There is therefore, in principle, a considerable amount of money that could be made available in the Nuclear Waste Fund for dry storage. However, under some circumstances, all these funds may eventually be required for the Yucca Mountain facility, whose total cost is projected to be $57.5 billion.[118] Furthermore, the use of the fund for interim storage has been blocked by utility lawsuits.[119] Most likely, therefore, the NWPA would have to be amended to allow the federal government to assume title to dry-stored spent fuel and responsibility for on-site storage.

An alternative approach would be to create an additional user fee similar to that which flows into the NWPA fund. A fee of 0.1 cents per nuclear kWh would generate an additional $750 million per year that could in 5 to 10 years

pay the $3.7 to 7 billion cost estimated above to transfer 35,000 tons of spent fuel into dry, hardened, on-site storage. Such a fee would, however, be opposed by the nuclear-plant operators.

## SUMMARY

As summarized in Table 2, we have proposed a number of possible actions to correct for the obvious vulnerabilities of spent fuel pools and to reduce the worst-case release that can occur from such pools. These recommendations would result in significant improvements over the current situation but they would also have significant limitations.

### Improvements

- The obvious vulnerabilities of spent fuel pools would be addressed.
- The worst-case release from a typical spent fuel pool of $^{137}$Cs—the isotope that governs the extent of long-term land contamination—would be reduced by a factor of about four. The residual inventory of $^{137}$Cs in the spent fuel pool would be about twice that in a reactor core.
- Our recommendations are achievable with existing technologies at a cost less than a percent of the price of nuclear-generated electricity.

### Limitations

- Considerable $^{137}$Cs would remain in hot spent fuel in pool storage.
- Terrorists could still cause releases from the dry-cask modules to which the aged spent fuel would be transferred, although it is difficult to imagine how they could release a large fraction of the total stored inventory, short of detonation of a nuclear weapon.
- Our analysis has been largely limited to accidents or terrorist acts that would partially or completely drain the pool while leaving the geometry of the spent fuel racks and the building above intact. Spent fuel fires might still arise in open-racked pools with air circulation blocked by a collapsed building. Such situations require more analysis.
- We have considered generic PWR pools. Additional issues may well arise when specific PWR and BWR pools designs are analyzed.

**Table 2:** Summary of proposals.

| Type | Action | Comment |
|---|---|---|
| **Regulation** | Congress should decide the probability of a terrorist-caused spent-fuel pool fire to be used by the NRC as a basis for regulatory cost-benefit analysis. | The NRC currently has no basis for deciding a limit on how much should be spent on strengthening protections against terrorist actions. |
| | The NRC should require that nuclear-power plant operators have the capability to operate and repair spent-fuel pools under accident conditions or after an attack. | This would apply the NRC's defense in depth approach for nuclear power plants to spent-fuel pools. |
| **Operation** | Minimize the movement of spent fuel casks over spent-fuel pools. | This has to be balanced with the proposal to remove older fuel from the pools. |
| | Minimize occasions when the entire core is moved to the pool during refueling outages. | Technically possible with some potential inconvenience to licensees. |
| | Transfer spent fuel to dry-cask storage 5 years after discharge from the power reactor. | Transfer probably could be accomplished somewhat earlier. Implementation will probably require Congress to permit use of the Nuclear Waste Fund or to enact a retrospective fee on electricity consumers—estimated at about 0.03–0.06 cents per kilowatt hour generated from the spent fuel. |
| **Design** | Return to open-frame storage—perhaps with additional measures of criticality control. | |
| | Provide for emergency ventilation of spent-fuel buildings. | Analysis is required on how to control this air supply if a fire did start. |
| | Install emergency water sprays. | Water from the sprays could block air circulation in a dense-packed pool or feed a fire under some circumstances. |
| | Make preparation for emergency repair of holes in pool walls and bottom. | |
| | Armor exposed outside walls and bottoms against projectiles. | Feasibility may vary greatly for different pool designs. |

Finally, all of our proposals require further detailed analysis and some would involve risk tradeoffs that also would have to be further analyzed. Ideally, these analyses could be embedded in an open process in which both analysts and policy makers can be held accountable. This process would have to be designed

to balance the need for democratic debate with the need to keep from general distribution information that might facilitate nuclear terrorism. We believe that our study shows that such a balance can be achieved.

## ACKNOWLEDGEMENTS

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## NOTES AND REFERENCES

1. "The results of the study indicate that the risk at SFPs [spent fuel pools] is low and well within the Commission's Quantitative Health Objectives. . . . The risk is low because of the very low likelihood of a zirconium fire even though the consequences of a zirconium fire could be serious." [*Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants* (NRC, NUREG-1738, 2001) Executive Summary].

2. *Spent Fuel Heatup Following Loss of Water During Storage* by Allan S. Benjamin et al. (Sandia National Laboratory, NUREG/CR-0649, SAND77-1371, 1979), fig. 14.

3. "Policy issues related to safeguards, insurance, and emergency preparedness regulations at decommissioning nuclear power plants storing fuel in spent fuel pools," (NRC, Secy-01-0100, June 4, 2001) pp. 3,5.

4. U.S. NRC, "In the matter of Dominion Nuclear Connecticut, Inc, (Millstone Nuclear Power Station, Unit No. 3)" Docket No. 50-423-LA-3, CLI-02-27, memorandum and order, Dec. 18, 2002.

5. *Ibid.*

6. *Ibid.*

7. Nuclear Waste Policy Act, 42 U.S.C. 10,131 et seq, Subtitle B.

8. *NRC's regulation of Davis-Besse regarding damage to the reactor vessel head* (Inspector General Report on Case No. 02-03S, Dec. 30, 2002, http://www.nrc.gov/reading-rm/doc-collections/insp-gen/2003/02-03s.pdf, accessed, Jan 4, 2003), p. 23.

9. Letter to the Senate majority and minority leaders, and Speaker and minority leader of the House of Representatives from the Attorneys General of Arizona, Arkansas, California, Colorado, Connecticut, Georgia, Hawaii, Iowa, Maryland, Massachusetts, Michigan, Minnesota, Mississippi, Montana, Nevada, New Jersey, New Mexico, New York, North Carolina, Ohio, Oregon, Pennsylvania, Rhode Island, Vermont, West Virginia, Washington and Wisconsin, Oct. 8, 2002.

10. List of spent-fuel pools from *Energy Resources International*, 2002, "2002 Summary of U.S. Generating Company In-Pool Spent Fuel Storage Capability Projected Year That Full Core Discharge Capability Is Lost," June, 2002, (http://www.nei.org/documents/Spent_Fuel_Storage_Status.pdf, Dec. 9, 2002). Latitudes and longitudes of the sites from http://geonames.usgs.gov/fips55.html.

11. In addition, Browns Ferry Unit 1 is nominally operational. However, it is defueled and not in service.

12. *Spent Nuclear Fuel Discharges from US Reactors 1994* (U.S. Department of Energy, Energy Information Agency, report # SR/CNEAF/96-0, 1996.

13. J. G. Ibarra, W. R. Jones, G. F. Lanik, H. L. Ornstein and S. V. Pullani, *Operating Experience Feedback Report: Assessment of Spent Fuel Cooling* (NRC, NUREG-1275, 1997), Vol. 12, figs. 2.1, 2.2.

14. See e.g. *Analysis of Spent Fuel Heatup Following Loss of Water in a Spent Fuel Pool: A Users' Manual for the Computer Code SHARP* by Energy and Environmental Science, Inc. (NUREG/CR-6441/ BNL-NUREG-52494, 2002).

15. Strontium-90 (28-year half-life) and its decay product, yittrium-90 (64 hours) account for another 40 percent of fission-product activity at 10 years [M. Benedict, T. H. Pigford, and H. W. Levi, *Nuclear Chemical Engineering*, *2nd ed.* (McGraw-Hill, 1981), Table 8.1]. However $^{90}$Sr is less volatile than $^{137}$Cs, especially under the oxidizing conditions typical of a spent fuel pool fire. It and $^{90}$Y are not gamma emitters and are therefore a hazard primarily if ingested.

16. *Table of Isotopes, 7th ed.*, C. M. Lederer and V. S. Shirley, eds. (John Wiley, 1978).

17. Exposures and effects of the Chernobyl accident," Annex J in *Sources and Effects of Ionizing Radiation* (UN, 2000) http://www.unscear.org/pdffiles/annexj.pdf, "Within these areas, radiation monitoring and preventive measures were taken that have been generally successful in maintaining annual effective doses within 5 mSv [0.5 rems]" ("Exposures and effects of the Chernobyl accident," pp. 472–5).

18. "Exposures and effects of the Chernobyl accident," p. 457.

19. Fission in LEU fuel yields 3.15 Curies of $^{137}$Cs per MWt-day of heat released. One Curie is the radioactivity of one gram of radium ($3.7 \times 10^{10}$ disintegrations/sec). 1 Bequerel (Bq) is one disintegration/sec.

20. Range estimated in *A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants* by R. J. Travis, R. E. Davis, E. J. Grove, and M.A. Azarm (Brookhaven National Laboratory, NUREG/CR-6451; BNL-NUREG-52498, 1997), Table 3.2. More detailed analysis is provided in *Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82* by V. L. Sailor, K. R. Perkins, J. R. Weeks and H.R. Connell (Brookhaven National Laboratory, NUREG/CR-4982 or BNL-NUREG-52083, 1987), Sections 3 and 4. Virtually all the $^{137}$Cs would be released from the spent fuel before the melting temperature of zirconium (1850°C) is reached. See "Report to the American Physical Society by the study group on radionuclide release from severe accidents at nuclear power plants," *Reviews of Modern Physics 57* (1985), p. S64. However, it is possible that some of the older fuel might not catch fire and some fraction of the $^{137}$Cs might plate out onto cool surfaces in the building.

21. For the "wedge model" the contamination level $\sigma = [Q/(\theta r R_d)]\exp(-r/R_d)$ Ci/m$^2$ where Q is the size of the release in Curies, $\theta$ is the angular width of a down-wind wedge within which the air concentration is assumed to be uniform across the wedge and vertically through the mixing layer, r is the downwind distance in meters, and $R_d$ is the "deposition length" $R_d = Hv_w/v_d$. H is the thickness of the mixing layer; $v_w$ is the wind velocity averaged over the mixing layer; and $v_d$, the aerosol deposition velocity, measures the ratio between the air concentration and ground deposition density. This "back-of-the-envelope" approximation was first used in the "Report to the American Physical Society by the study group on light-water reactor safety," *Reviews of Modern Physics*, 47, Supplement 1 (1975), p. S97. For a uniform population density, the population radiation dose is independent of $\theta$. An extensive discussion of aerosol formation and deposition

may be found in "Report to the American Physical Society by the study group on radionuclide release from severe accidents at nuclear power plants," p. S69–S89. Data on the frequency of different dispersion conditions in the U.S. and data on aerosol deposition rates may be found in *Reactor Safety Study*, (U.S. NRC, NUREG-75/014, 1975), Appendix VI-A. See also: *Probabilistic Accident Consequence Uncertainty Analysis: Dispersion & Deposition Uncertainty Assessment*, (U.S. Nuclear Regulatory Commission & Commission of European Communities, NUREG-6244 and EUR 15855EN, 1995), Vols. 1–3.

22. D. I. Chanin and M. L. Young, *Code Manual for MACCS2: Volume 1, User's Guide*, Sandia National Laboratories, Albuquerque, NM, SAND97-0594, March 1997. In the Gaussian plume model with a mixing layer thickness H and a constant wind velocity $v_w$, the time-integrated plume concentration at a point on the ground a horizontal distance y from the centerline of the plume and a distance h below it is $\chi = [Q/(\pi\sigma_y\sigma_z v_w)]$ $\exp[-y^2/(2\sigma_y^2)]$ $\{\exp[-h^2/(2\sigma_z^2)] + \sum_{n=1-\infty}[\exp[-(2nH-h)^2/(2\sigma_z^2)] + \exp[-(2nH+h)^2/(2\sigma_z^2)]]\}$. The term $\sum_{n=1-\infty}[\exp[-(2nH-h)^2/(2\sigma_z^2)] + \exp[-(2nH+h)^2/(2\sigma_z^2)]]$ takes into account multiple reflections of the plume off the top of the mixing layer and the ground. Q, $\sigma_y$, and $\sigma_z$ are all functions of downwind distance. Q, the number of Curies in the plume, is reduced by deposition. The area deposition concentration is $v_d\chi$, where $v_d$ is the deposition velocity.

23. The calculations used the same median values of mixing layer height (1000 m), wind velocity (5 m/sec), and deposition velocity (0.01 m/sec) used in the wedge-model calculation above. On the basis of a match with the wedge-model value $\theta r = 2.4\ \sigma_y = 11$ km at r = 100 km downwind, dispersion conditions have been chosen to be Pasquill D-type which the MACCS2 code parameterizes as $\sigma_y = 0.1474x^{0.9031}$ and $\sigma_z = 0.3x^{0.6532}$ m where x is the downwind distance in meters.

24. The heat of combustion of zirconium is 8.7 and 4.1 million j/kg in air and steam respectively. We assume that the pool contains 80 tons of zirconium, i.e., 0.2 tons per ton of U.

25. Most of the data in the charts are from 1998 data provided by utility companies to the NRC and previously displayed on its web site at http://www.nrc.gov/OPA/drycask/sfdata.htm. Post September 11, 2001, such data are no longer available on the web. The storage capacity in the storage pools of a few plants has increased since 1998 due to reracking with higher density racks. Such increases are included for the following reactors: Crystal River 3 ["Florida Power Corporation, Crystal River Unit 3, Environmental Assessment and Finding of No Significance" (NRC, *Federal Register* (FR), v. 65, n. 177, pp. 55059–55061, Sept. 12, 2000)]; Callaway [*FR*, v. 64, n. 10, pp. 2687–2688, Jan. 15, 1999]; Nine Mile Point 1 [*FR*, v. 64, n. 70, pp. 18059–18062, April 13, 1999]; and Kewaunee [*FR*, v. 65, n. 236, pp. 76672–76675, Dec. 7, 2000]. Three other plants (Enrico Fermi 1, Comanche Peak, and Vermont Yankee) have re-racked, but no capacity data are available (no environmental assessments were done for them). Brunswick 1 and 2 and Robinson are shipping spent fuel to the Harris plant, also in North Carolina and owned by Carolina Light and Power Company. Nine Mile Point 2, Pilgrim 1, Summer, and Three Mile Island 1 plants intend to re-rack their spent fuel in the next few years ("2002 Summary of U.S. Generating Company In-Pool Spent Fuel Storage Capability Projected Year That Full Core Discharge Capability Is Lost"). Big Rock Point, Browns Ferry 3, Diablo Canyon 1&2, Duane Arnold, Farley 1&2, Grand Gulf 1, Haddam Neck, Humboldt Bay, Palo Verde 1–3, River Bend 1, San Onofre 1–3, Sequoyah 1&2, Washington Nuclear, and Yankee Rowe plants, some of which are being decommissioned, all intend to add dry storage in the next few years (*ibid*). An

earlier version of this figure appeared in Allison Macfarlane, "Interim storage of spent fuel in the United States," *Annual Review of Energy and the Environment* 26 (2001), pp. 201–235.

26. "Simulation of the Chernobyl dispersion with a 3-D hemispheric tracer model" by Janusz Pudykiewicz, *Tellus 41B* (1989), pp. 391–412.

27. "Exposures and effects of the Chernobyl accident," Table 8.

28. One rem = 0.01 Sievert. For estimated exposure-dose coefficients, see *Ionizing Radiation: Sources and Biological Effects* (UN, 1982), Annex E, Table 27 (external) and Table 33 (ratio of internal to external). For the external dose, the $^{137}$Cs is assumed to have weathered into the soil with an exponential profile with a mean depth of 3 cm. Shielding by buildings is estimated to reduce the dose by a factor of 0.4 for wooden homes and 0.2 for masonary homes. The resulting total dose-reduction is by a factor of about 1/6. Self shielding by the body is assumed to reduce the dose by an additional average factor of 0.7. See also *Federal Guidance Report No. 12: External Exposure To Radionuclides In Air, Water, And Soil* by K. F. Eckerman and J. C. Ryman (Oak Ridge National Laboratory, EPA-402-R-93-081, 1993) Table II-6. The additional cancer death risk was assumed to be 1/1700 per rem, including a recommended reduction factor of 2 for the risk of chronic radiation per rem relative to that from an "acute" (instantaneous) dose such as that at Hiroshima and Nagasaki ["Epidemiological Evaluation of Radiation-Induced Cancer," Annex I in *Sources and Effects of Ionizing Radiation* (UN, 2000), p. 361.] Note that arguments about the validity of a linear extrapolation to low doses from the high doses at which epidemiological evidence is available are irrelevant in this dose range. The mean dose among the cohort of Hiroshima-Nagasaki survivors who have been followed in Life-Span Study is 21 rem (*op. cit.*, Table 6). A statistically significant response has been found down to 5 rem for solid cancers with a cancer dose-effect response for solid cancers linear up to about 300 rem ["Studies of the mortality of atomic bomb survivors, Report 12, Part I. Cancer: 1950–1990" by D. A. Pierce, Y. Shimizu et al. *Radiation Research 146* (1), p. 10, 1996.]

29. *A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants.* The value of the agricultural land was assumed to be \$0.2 million/km$^2$. The value of the condemned land would therefore be \$0.4–1.4 billion. The remainder of the cost was assumed to be \$0.074 million per permanent evacuee. Therefore, 1.6–7.6 million people would be permanently evacuated in this scenario. \$17–279 billion of these consequences were assumed to occur beyond 50 miles where the population density was assumed to be 80/km$^2$. This would correspond to an evacuated area beyond 50 miles of 1100–19,000 km$^2$. We have done a calculation using the MACCS2 code to obtain, for 3.5–35 MCi $^{137}$Cs releases with 40 MWt plume heat, damage estimates of \$50–700 billion plus 50,000–250,000 cancer deaths among people remaining on contaminated land [2000 person-rem per cancer death, valued in NRC cost-benefit analyses at \$4 million per cancer death, (Nuclear Regulatory Commission, *Regulatory Analysis Technical Evaluation Handbook* NUREG/BR-0184, 1997)]. An average population density of 250/km$^2$ was assumed (population density of the U.S. Northeast). Evacuation was assumed if the projected radiation dose was greater than 0.5 rems per year (EPA Protective Action Guide recommendation). The losses due to evacuation were assumed to be \$140,000/person for fixed assets, \$7,500/person relocation costs, and \$2,500/hectare for farmland abandoned because of the projected contamination level of its produce. Two possible decontamination factors (DF) were assumed: DF = 3 and 8 at costs of \$9,000 and \$20,000 per hectare of farmland (assumed to be 20% of the total area) and \$19,000 and \$42,000 per resident (value for a "mixed-use" urban area), excluding

the cost of disposal of the radioactive waste [based on D.I. Chanin and W.B. Murfin, *Estimation of Attributable Costs from Plutonium Dispersal Accidents* (Sandia National Laboratory, SAND96-0957, 1996)]. Based on these cost assumptions, no farmland would be decontaminated but decontamination would be performed in residential areas up to contamination levels that prior to decontamination would result in doses of 4 rems per year up to the end of temporary relocation periods that are assumed to last up to 30 years. The range of $^{137}$Cs contamination levels in areas where decontamination would be carried out is from about 2.5 up to 80 Ci/km$^2$.

30. Calculated using the Origin 2.1 computer code [*ORIGEN 2.1: Isotope Generation and Depletion Code Matrix Exponential Method*, CCC-371 ORIGEN 2.1, (Oak Ridge National Laboratory, Radiation Safety Information Computational Center, August 1996)].

31. In 1996, the NRC staff reported an example in which boiling would occur in 8 hours instead of 4.5 days because the core had been loaded into the spent fuel pool 5 days after shutdown instead of 23 in a previous refueling at the same reactor (NRC, "Briefing On Spent Fuel Pool Study," Public Meeting, November 14, 1996, http://www.nrc.gov/reading-rm/doc-collections/commission/tr/1996/19961114a.html, accessed Dec. 10, 2002, p. 27). This is consistent with the following calculation: Assume a generic PWR pool with an area of 61.3 m$^2$ and depth of 11.5 m containing about 600 metric tons of water, as described in *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, p. A1A-2. [A more detailed calculation would take into account the amount of water displaced by the fuel assemblies. In subsequent calculations, we will assume 471 kg U per fuel assembly with cross-section of 21.4 $\times$ 21.4 cm and a height of 4 meters. Such an assembly has 59% water content by volume (*Nuclear Engineering International*, September 2001, p. 24).] For a pool inventory of 340 tons of 1–20 year-old fuel generating an average decay heat of 3 kWt/tU with or without a freshly discharged core containing 85 metric tons of uranium generating 120 kWt/tU decay heat 4 days after shutdown, the total decay heat would be 1 or 11 MWt. Given the heat capacity of water of 4200 joules/kg-°C, the decay heat would raise the temperature of the pool from 30 to 100°C in 4.4 or 50 hours and thereafter boil off 0.026 or 0.29 meters of water per hour (the latent heat of vaporization of water is 2.3 Mj/kg). Assuming that there are 7 meters of water above the fuel, it would take 1 or 11 days before the radiation shield provided by the water covering was reduced to 1 meter.

32. In principle, removing the spent fuel assemblies and reshuffling the rest before inserting fresh fuel should be faster. However, any departure from a choreographed reshuffle (due, for example, to discovery of damaged fuel) requires time-consuming recalculation of the subcriticality margin (David Lochbaum, Union of Concerned Scientists, private communication, Jan. 7, 2003).

33. "NRR [Nuclear Reactor Regulation staff] determined through a recent survey of all power reactors . . . that some sites do not have anti-siphon devices in potential siphon paths. During refueling operations . . . a flow path exists to the reactor vessel, inventory loss [could occur] through the RHR (residual heat removal), chemical and volume control system, or reactor cavity drains [or the] shipping cask pool drains. For these situations in many designs, the extent of the inventory loss is limited by internal weirs or internal drain path elevations, which maintain the water level above the top of the stored fuel . . . During the NRR survey assessment, the staff found that five SFPs (spent fuel pools) have fuel transfer tubes that are lower than the top of the stored fuel without interposing structures." (*Operating Experience Feedback Report: Assessment of Spent Fuel Cooling*, NUREG-1275, pp. 5–6). In 1994, about 55,000 gallons [200 m$^3$] of water leaked from piping, which had frozen in an unheated containment fuel pool transfer system

at the closed Dresden I station. The NRC noted the potential for a "failure of 42"[inch, 1 m] fuel transfer tube [which] could rapidly drain fuel pool to a level several feet [>1 m] below top of [660] stored fuel bundles." [*Dresden, Unit 1 Cold Weather Impact on Decommissioned Reactor (Update)*, U.S. NRC, January 24, 1994, pp. 94–109].

34. *Operating Experience Feedback Report: Assessment of Spent Fuel Cooling*, NUREG-1275, p. 32 and Fig. 3.2.

35. Doses calculated from a dry pool containing 650 tons of 43 MWd/kgU spent fuel in a square array with 1.4 cm pitch. The fuel is a composite with a mix of the following cooling times: 20 tons each at 30 days, 1 year, and 2 years; 100 tons at 5 years; 240 tons at 10 years; and 250 tons at 25 years. The gamma-ray source intensities within the fuel were calculated using ORIGEN2, grouped in 18 energy intervals. These radiation-source data were then used as input to the MCNP4B2 code [*Los Alamos National Laboratory, Monte Carlo N-Particle Transport Code System* (Radiation Safety Information Computational Center, CCC-660 MCNP4B2 1998)] which was used to perform radiation transport calculations to obtain the flux and energy spectra of the gamma-rays 1 m above the floor of the building at radii of 5, 10 and 15 meters from its center. The radiation doses were then calculated using the "American National Standard for Neutron and Gamma-Ray Fluence-to-Dose Factors" (American Nuclear Society, ANSI/ANS-6.1.1, 1991) and an average self-shielding factor of 0.7. The concrete has a density of 2.25 gms/cc and a composition in weight percent of 77.5% $SiO_2$, 6.5% $Al_2O_3$, 6.1% CaO, 4.0% $H_2O$, 2.0% $Fe_2O_3$, 1.7% $Na_2O$, 1.5% $K_2O$ 0.7% MgO ("Los Alamos concrete, MCNP4B2 manual, pp. 5–12). In the absence of a roof, the dose rates at 10 and 15 meters would be reduced by factors of 0.37 and 0.24 respectively. Similar calculations for 400 tons of 33MWd/kgU spent fuel (25% each 30-day, 1-yr, 2-yr and 3-yr cooling) reported in *Spent Fuel Heatup Following Loss of Water During Storage*, Appendix C: "Radiation dose from a drained spent-fuel pool" give a dose rate of about 300 rads/hr at ground level 15 m from the center of a rectangular 10.6 × 8.3 m pool.

36. Among the emergency workers at Chernobyl, deaths began for doses above 220 rems. The death rate was one third for workers who had received doses in the 420–620 rem range and 95% (1 survivor) for workers who received higher doses ("Exposures and effects of the Chernobyl accident," Table 11).

37. *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, p. A1A-1.

38. Figure 5 was calculated with ORIGEN 2.1 assuming that the initial enrichments for burnups of 33, 43, 53 and 63 MWd/kgU were 3.2, 3.7, 4.4 and 5.2% respectively. The PWRU.LIB and PERU50.LIB cross-section files were used to calculate the production rates of actinides and fission products in PWR fuel.

39. S. R. Tieszen, *Fuel Dispersal Modeling for Aircraft-Runway Impact Scenarios* (Sandia National Laboratory, SAND95-2529, 1995), p. 73.

40. *Fuel Dispersal Modeling for Aircraft-Runway Impact Scenarios*, p. 70.

41. *World Trade Center Building Performance Study*, (FEMA, 2002) Appendix E, http://www.fema.gov/library/wtcstudy.shtm accessed Dec. 10, 2002.

42. On May 16, 1979, the government of the German state of Lower Saxony issued a ruling about a proposed nuclear fuel center at Gorleben. One aspect of the ruling was a refusal to license high-density pool storage, in part from concern about war impacts. The ruling followed a public hearing in which more than 60 scientists, including two of the present authors (J. B. and G. T.) presented their analyses. A third author (K. J.) had been

responsible for the design of the pool and subsequently oversaw the design of the dry casks currently used in Germany [Klaus Janberg, "History and actual status of aircraft impact and anti-tank weaponry consequences on spent fuel storage installations," paper presented at the International Conference on Irradiated Nuclear Fuel, Moscow IFEM, September 11, 2002]. A brief description (in German) and photographs and diagrams of the German dry-cask central storage facility that was built at Gorleben instead of a spent-fuel pool may be found in *Brennelementlager Gorleben, BLG*, http://www.math.uni-hamburg.de/math/ign/hh/1fi/blg.htm, accessed Dec. 10, 2002. A similar dry-cask storage facility was built instead of a storage pool at Ahaus, Germany.

43. Swiss Federal Nuclear Safety Inspectorate (HSK), Memorandum, "Protecting Swiss Nuclear Power Plants Against Airplane Crash" (undated), p. 7. This memo also describes Swiss protection requirements (the same as those in Germany) http://www.hsk.psi.ch/pub_eng/publications/other%20publications/2001/AN-4111_E-Uebersetz_Flz-absturz.pdf accessed, Jan. 9, 2003.

44. "In estimating . . . catastrophic PWR spent fuel pool damage from an aircraft crash (i.e., the pool is so damaged that it rapidly drains and cannot be refilled from either onsite or offsite resources), the staff uses the point target area model and assumes a direct hit on a 100 × 50 foot spent fuel pool. Based on studies in NUREG/CR-5042, *Evaluation of External Hazards to Nuclear Power Plants in the United States*, it is estimated that 1 of 2 aircrafts are large enough to penetrate a 5-foot-thick reinforced concrete wall . . . It is further estimated that 1 of 2 crashes damage the spent fuel pool enough to uncover the stored fuel (for example, 50 percent of the time the location is above the height of the stored fuel)" (*Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, p. 3–23).

45. See e.g. *Accident Analysis for Aircraft Crash into Hazardous Facilities* (U.S. Department of Energy, DOE-STD-3014-96, 1996), Appendix C. We have used these formulae for an aircraft turbine shaft weighing 400 kg with a diameter of 15 cm and traveling at 156 m/sec (350 miles per hour, speed of the aircraft that crashed into the Pentagon according to NEI, see following footnote) and 260 m/sec [590 miles/hr, estimated speed of the aircraft that crashed into the World Trade Center South Tower, (*World Trade Center Building Performance Study*)]. They predict that such an object could perforate a reinforced concrete wall 0.8 to 1.8 meters thick, depending primarily on the impact speed.

It is possible that a spent-fuel pool, with its content of water mixed with dense fuel assemblies, might resist penetration more like an infinitely thick slab. In this case, the range of penetration depths for the large aircraft turbine shaft becomes 0.4–1.3 m. For a useful review, which shows the great uncertainty of empirical penetration formulae and the very limited ranges over which they have been tested empirically, see *Review of empirical equations for missile impact effects on concrete* by Jan A. Teland (Norwegian Defense Research Establishment, FFI/RAPPORT-97/05856, 1998).

An additional reference point is provided by the NRC staff's conclusion that "if the cask were dropped on the SFP [spent-fuel-pool] floor, the likelihood of loss-of-inventory given the drop is 1.0" (*Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, p. A2C-3). For a drop height of 12 m (the depth of a pool) the kinetic energy of a 100-ton cask (neglecting the absorption of energy by displacing water and crushing spent-fuel racks) is about $10^7$ joules—about the same as the energy of the large jet turbine shaft at a velocity of about 240 m/sec. Because of the larger hole that the cask would have to punch, the energy absorbed by the structure would be expected to be larger. It should also be noted that the weight of the entire jet engine is about 4,000 kg, its diameter, including the fan blades, is about

the same as a spent-fuel cask and its kinetic energy at 240 m/sec is about 10 times greater.

46. *Aircraft crash impact analyses demonstrate nuclear power plant's structural strength* (Nuclear Energy Institute Press release, Dec. 2002, http://www.nei.org/documents/EPRINuclearPlantStructuralStudy200212.pdf, accessed Jan. 5, 2003).

47. *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, p. A2C-3.

48. *Analysis of the Total System Lifecycle Cost of the Civilian Radioactive Waste Management Program*, (U.S. DoE, Office of Civilian Waste management, Report # DOE/RW-0533, 2001), pp. 1–7.

49. "Nuclear Waste: Uncertainties about the Yucca Mountain Repository Project," testimony by Gary Jones, Director, Natural Resources and Environment, U.S. General Accounting Office, before the Subcommittee on Energy and Air Quality, House Committee on Energy and Commerce, 21 March 2002.

50. Charles Pennington, NAC International, private communication, Dec. 2, 2002.

51. In recently installed racks, the boron is contained in Boral sheets composed of boron carbide ($B_4C$) in an aluminum matrix, permanently bonded in a sandwich between aluminum plates. This design has proven more durable than a previous design in which boron carbide was mixed 50 percent by volume with carbon, formed into a 1/4-inch thick sheet and clad in 1/8-inch stainless steel (*Spent Fuel Heatup Following Loss of Water During Storage*, p. 19).

52. A vendor's representation of dense-pack fuel racks is available at http://www.holtecinternational.com

53. This problem could be mitigated to some degree by putting holes in the walls of the dense-pack racks—subject to limitation that considerable neutron absorption in the walls is required keep the spent fuel subcritical. The holes would allow air to circulate through the racks above the water surface. The 1979 Sandia report concluded that such an approach could be effective for fuel a year or more old (*Spent Fuel Heatup Following Loss of Water During Storage*, pp. 78).

54. Based on heat capacities of $UO_2$ and Zr of 0.3 joules/gmU-°C [S. Glasstone and A. Sesonske, *Nuclear Reactor Engineering* (Van Nostrand Reinhold, 1967) Table A7] and assuming 0.2 grams of Zr per gram U, the heat capacity of reactor fuel is about 0.4 joules/gmU-°C. In a 1997 study done by Brookhaven National Laboratory for the NRC, the "critical cladding temperature" was chosen as 565°C. This was the temperature for "incipient clad failure" chosen in the previous Workshop on Transport Accident Scenarios where "expected failure" was fixed at 671°C. The Brookhaven group chose the lower temperature for fuel failure in a spent-fuel-pool drainage accident because "it would take a prolonged period of time to retrieve the fuel, repair the spent fuel pool or establish an alternate means of long-term storage" [*A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants*, pp. 3–4.]

55. The gas-diffusion-limited zirconium oxidization rate has been parameterized as $dw^2/dt = K_o\exp(-E_a/RT)$ in the range 920–1155°C, where $w$ is the weight gain of the cladding (g/cm$^2$) due to oxidation, $K_0$ is the rate constant [$5.76 \times 10^4$ (gm/cm$^2$)$^2$/sec], $E_a$ is the activation energy (52990 calories), R is the gas constant (1.987 cal/°K), and T is the absolute temperature (°K) (*Spent Fuel Heatup Following Loss of Water During Storage*, p. 31–34). At 920°C, therefore, $K_o\exp(-E_a/RT) = 1.1 \times 10^{-5}$ (gm/cm$^2$)$^2$/sec. The

fuel cladding contains 0.34 gmZr/cm$^2$. w$^2$ for full oxidation to $ZrO_2$ will therefore be about 0.014 (gm/cm$^2$)$^2$. Thus, the characteristic time for complete oxidation would be about 15 minutes at 920°C and would decrease rapidly as the temperature increased further.

The Advisory Committee on Reactor Safeguards (ACRS) has raised the possibility that, for high-burnup fuel, the ignition temperature might be considerably lower: "there were issues associated with the formation of zirconium-hydride precipitates in the cladding of fuel especially when the fuel has been taken to high burnups. Many metal hydrides are spontaneously combustible in air. Spontaneous combustion of zirconium-hydrides would render moot the issue of 'ignition' temperature ..." In addition, the ACRS points out that nitrogen reacts exothermically with zirconium, "[this] may well explain the well-known tendency of zirconium to undergo breakaway oxidation in air whereas no such tendency is encountered in either steam or in pure oxygen" ["Draft Final Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants," letter from Dana Powers, ACRS chairman, to NRC Chairman Meserve, April 13, 2000, p. 3].

56. *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, "Executive Summary," p. x.

57. Between 300 and 1200 °K, the longitudinal conductivity of a 0.4-cm radius rod of $UO_2$ clad in zircalloy with an inside radius of 0.41 cm and a cladding thickness of 0.057 cm is about k = 0.06 Watts/(°C/cm) [based on temperature-dependent conductivities for $UO_2$ falling from 0.076 to 0.03 and for zircalloy rising from 0.13 to 0.25 Watts/[cm$^2$-(°C/cm)] (International Nuclear Safety Center, http://www.insc.anl.gov/matprop/uo2/cond/solid/thcsuo2.pdf, Table 1; http://www.insc.anl.gov/matprop/zircaloy/zirck.pdf, Table 1, accessed Dec. 19, 2002)]. The density of uranium in the $UO_2$ is about 10 gm/cc. A rod 400 cm long would therefore contain about 2 kg of uranium. For a fuel rod L cm long containing M kg U and cooled at both ends to a temperature $T_0$, with a heat generation rate of P Watts/kgU uniformly distributed along its length, the temperature difference between the center and ends would be PML/(8k) $\approx$ 1700 P °C. Taking into account the thermal conductivity of the steel boxes and boral surrounding the fuel assemblies in the dense-pack configuration lowers this estimated temperature increase to approximately 1000 P °C.

58. Within the fuel assembly, the net radiation flux in the z direction is approximately $F = -4f\sigma T^3(dT/dz)\langle\lambda_z\rangle$ where f is the fraction of the area of the fuel assembly between the fuel rods (about 0.6) and $\langle\lambda_z\rangle = \int d\Omega(\mathrm{Cos}\theta)[\lambda(\theta,\phi)]$ is the average distance that radiation travels up the fuel assembly before being reabsorbed—on the order of centimeters. We have made the approximation that the difference in temperature between the radiating and absorbing points can be calculated using the first derivative of T. We also assume that the rate of heat generation is constant at a rate of PM/(AL) Watts/cm$^3$ along the length (L = 400 cm) of the fuel assembly. In this approximation, the temperature profile can be calculated as $T = [1000PM/(A\sigma)]\{[-(z/L) - z^2/(2L^2)]L/(f\langle\lambda_z\rangle) + 1\}^{1/4}$°K, where z is negative and measured in centimeters downward from the top of the fuel assembly. When z = −L, $T(-L) = 600\{P[1 + (0.8L/\langle\lambda_z\rangle)]\}^{1/4}$°K. For P = 1 kW/tU, T(−L) = 2300 or 1700°C if $\langle\lambda_z\rangle$ = 1 or 3 cm respectively.

59. Assume that a fuel rod has a length L, contains M = 2 kg of uranium, generates decay heat at a rate of P watts/kgU, has a temperature $T_{max}$ at its top and that the water level is at $z_w$ m (where z = 0 is the bottom of the fuel). In the approximation where the heat rate along the length of the fuel is constant, the combined rate of input of heat into the water from the submerged part of the fuel and from black body radiation impinging on the water's surface will be $P_- = PMz_w/L + P_{bb-}$. The heat generation rate of the

fuel above the water will be $P_+ = PM(L - z_w)/L$. The cooling of the above-water fuel is limited, however, by the availability of steam generated by the below-water fuel. The rate of steam generation will be $P_-/2300$ grams/sec. When z falls below the bottom of the fuel assembly, $P_- = P_{bb-}$. We approximate $P_{bb-} = (A/264)\sigma(T_0 + 273)^4$ where $(A/264) = 2\ cm^2$ is the area in a fuel-assembly box for each of the 264 fuel rods and $T_0$ is the temperature at the bottom of the fuel assembly. In *Spent Fuel Heatup Following Loss of Water During Storage*, Fig. B-1, it is estimated that $T_0 = 200°C$ at the point when $T_{max} = 900°C$, i.e., when the fuel is about to fail. This gives $P_{bb-} \approx 0.6$ Watts. Assuming perfect heat transfer, the steam will heat to a temperature $T_{max}°C$ as it passes through the fuel assembly and absorb approximately $2.1(T_{max}$ -100) joules per gram. Therefore, in order to remove the power $P_+$ and maintain the above water fuel in equilibrium, it is necessary that $P_+ < 2.1(T_{max} - 100)P_{bb-}/2300\ M \approx 0.3$ Watts/kgU when $T_{max} = 1200°C$. This means that the fuel has to be about 100 years old after discharge before steam cooling will remain effective when the water level drops to the bottom of the fuel assembly.

60. For information on the strength of steel at high temperatures, see http://www.avestapolarit.com/template/Page2171.asp, accessed Jan. 10, 2003. The zircaloy tubes of a Canadian CANDU reactor slumped at 1200°C (see *CANDU Safety # 17—Severe Core Damage Accidents*, V. G. Snell, Director Safety & Licensing, http://engphys.mcmaster.ca/canteach/techdoclib/CTTD-0014/CTTD-0014-17/17of25.pdf, accessed Jan 10, 2003).

61. For a square box with inside dimensions of 0.225 m containing a fuel assembly with 264 rods with diameters of 0.95 cm, [*Analysis of Spent Fuel Heatup Following Loss of Water in a Spent Fuel Pool: A Users' Manual for the Computer Code SHARP*, Tables 2.1 and 2.2].

62. This can be derived from the gas momentum conservation equation, $\partial(\rho v)/\partial t + \partial(\rho v^2)/\partial z + P_L = -\partial P/\partial z - \rho g$ where $\rho$ is the air density, v is its velocity, P is the pressure, $P_L$ represents the pressure loss due to friction in the channel and $g = 10\ m/sec^2$ is the gravitational constant. For an equilibrium situation, the first term disappears. Integrating from the bottom of the spent fuel (z = 0) to its top (z = L = 4 m) gives $\rho_L(v_L)^2 - \rho_0(v_0)^2 + \int_0^L P_L dz = P(0) - P(L) - g\int_0^L \rho dz$. Assuming that: the pressure is constant across the top and bottom of the spent fuel, the gas velocity is constant below the spent fuel, the air velocity is zero at the top of the down-comer, and neglecting friction losses in the down-comer and beneath the spent fuel, we may subtract the momentum conservation equation for the down-comer (dc) from that for the fuel assembly (fa) and obtain $\rho_L(v_L)^2 + \int_0^L P_L dz = g\int_0^L[\rho_{dc} - \rho_{fa}]dz$. As indicated in the text, we approximate $\rho_0 = 1\ kg/m^3$, $\int_0^L \rho_{dc} dz \approx L\rho_0$, and $\int_0^L \rho_{fa} dz \approx 0.5\ L\rho_0$. This gives $\rho_L(v_L)^2 + \int_0^L P_L dz \approx 0.5\ g\rho_0 L = 20$ joules/$m^3$. Noting that $\partial(\rho v)/\partial z$ is a constant and that, at constant pressure, $\rho \sim T^{-1}$, where T is the absolute temperature, $\rho_L(v_L)^2 = \rho_0(v_0)^2(T_L/T_0)$, where $T_L = 1173\ °K$ at the ignition point. We assume that $T_0 = 100\ °C = 373\ °K$. We then obtain $3.1(v_0)^2 + \int_0^L P_L dz = 20$ joules/$m^3$ and $v_0 \approx 2.5$ m/s, if the $P_L$ term is neglected.

$P_L$ may be approximated as the sum of a loss term due to the constriction of the air passing through the base-plate hole and surface friction within the fuel assembly, $\int_0^L P_L dz = K_0\rho_0(v_0)^2 + \int_0^L f\rho v^2 dz/(2D_H)$. Here $K_0 = 2(1-x)/x$, $x = (A_h/A_f)^2$, $A_h$ is the area of the hole in the base-plate and $A_f = S^2 - 264\ \pi(D/2)^2$ is the cross-sectional area of the air flow inside the box around the fuel assembly. (S = 0.225 m is the inside width of the box and D = 0.0095 m is the outside fuel-rod diameter). For a dense-pack arrangement with a 5 inch [13 cm] hole in the base-plate, $x \approx 0.15$ and $K_0 \approx 11.3$. In the second pressure-loss term, L = 4m is the height of the fuel assembly, f is the friction factor, $D_H = 4\ A_f/P_w$ is the "hydraulic diameter" of the channel, and $P_w = 4S + 264\ \pi D$ is the total perimeter

of all the surfaces in the cross-section (*Users' Manual for the Computer Code SHARP*, pp. 4–7, 4–16). For the fuel assembly in our example, $D_H \approx 0.015$ m. The friction factor may be written as $f = C/(Re)^n$, where $Re = \rho v\ D_H/\mu$ is the Reynolds number, and $\mu$ is the viscosity of air ($31 \times 10^{-6}$ pascal-seconds at 600°K). The exponent $n = 1$ for laminar flow ($Re < 2100$), which will be seen to be the case in the fuel assembly. The coefficient $C \sim 100$ within the fuel assembly in the approximation where all rods are treated as interior rods (*ibid.*, p. 4–7, 4–16/17). Thus, $\int_0^L P_L dz = K_0\rho_0(v_0)^2 + \{C\mu/[2(D_H)^2]\} \int_0^L v dz \approx K_0\rho_0(v_0)^2 + 55v_0$ joules/m$^3$, where we have approximated $\int_0^L v dz \approx 2Lv_0$, where $v_0$ is the entrance velocity to the air at the base of the fuel assembly. If we add this friction pressure term to the equation at the end of the paragraph above, we get $14.4(v_0)^2 + 55\ v_0 = 20$ joules/m$^3$ or $v_0 \approx 0.33$ m/sec.

An approximation of open-rack storage could be obtained by dropping the base-plate constriction term (i.e., setting $x = 1$) and dropping the S in the perimeter term above. Then, if the center-to-center spacing of the fuel assemblies is increased by a factor of $5^{1/2}$ in going from dense-pack to an open-array spacing with a fuel-assembly density lower by a factor of five, $D_H \approx 0.1$ m and the equation above becomes $3.1(v_0)^2 + 1.24\ v_0 = 20$ joules/m$^3$, or $v_0 = 2.3$ m/sec, which would make it possible to cool a pool filled with fuel generating about 100 KWt/tU. If the hot fuel were surrounded by cooler fuel assemblies, cross flow from the cooler to the hot assemblies would provide still more cooling.

63. *Users' Manual for the Computer Code SHARP*, Figs. 6.3 and 6.5. Our result obtained in the previous footnote corresponds to the case for a wide (e.g., 8-inch or 20 cm) downcomer and constant room temperature.

64. *Spent Fuel Heatup Following Loss of Water During Storage*, fig. 3, p. 85.

65. The 2001 *Users' Manual for the Computer Code SHARP* notes the availability of only "limited data [from] one experiment . . . in a three parallel channel setup" (p. 5-1).

66. *Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82* by V. L. Sailor, K. R. Perkins, J. R. Weeks, and H. R. Connell (Brookhaven National Laboratory, NUREG/CR-4982; BNL-NUREG-52093, 1987), p. 52.

67. *Op cit,* pp. 52, 53, 63.

68. Complete blockage would, however, tend to quench the fire.

69. See, for example: J. H. Jo, P. F. Rose, S. D. Unwin, V. L. Sailor, K. R. Perkins and A. G. Tingle, *Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools* (Brookhaven National Laboratory, NUREG/CR-5281, 1989). Measures discussed and rejected because of perceived lack of cost-benefit included low density storage and water sprays. Management recommendations to reduce risk have been considered in, *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants.*

70. To compute the 0.7 and 5 percent probabilities, we compared an investment of \$5 billion in dry storage casks (midpoint of our estimated \$3.5–7 billion cost range) with a range of estimated costs for spent fuel fires. In footnote 29 the median damages (including cancer deaths at \$4 million each) from a 10–100 percent release of $^{137}$Cs from 400 tons of spent fuel are estimated at \$250–1700 billion. We discount these damages to \$100–750 billion because the risk would not be completely eliminated by the measures that we propose and their mitigating effect could occur decades after the investment. The $0.6 - 2.4 \times 10^{-6}$ probability of a spent-fuel fire per pool-year estimated in *Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants* (Table 3.1)

is equivalent to about 0.6 percent in 30 years for the 103 operating power reactors in the U.S.

71. *Spent Fuel Heatup Following Loss of Water During Storage*, "Conclusions," p. 85.

72. *Operating Experience Feedback Report, Assessment of Spent Fuel Cooling*, NUREG-1275, Vol. 12, p. 27.

73. Further discussion of defense in depth is provided in *Robust Storage of Spent Nuclear Fuel* by Gordon Thompson (Institute for Resource and Security Studies, Cambridge, MA, January 2003).

74. *Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants*, pp. 3–16 and Appendix 2C p. A2C-3 and –4.

75. Above, it was noted that an important motivation for moving the entire core into the spent-fuel pool was the need to recalculate the subcriticality of the core in the reactor pressure vessel if there are unplanned fuel movements. This problem deserves a separate study of its own.

76. David Lochbaum, Union of Concerned Scientists, private communication, Jan. 9, 2003.

77. Assuming a thermal to electric power conversion efficiency of one third, an 85 percent capacity factor, and a fuel burnup of 47 MWd/kg. The Sandia study considered fuel with a burnup of only 33 MWd/kgU. However, as can be seen from Figure 5, the decay heat at short decay times (less than a year or so) is insensitive to the fuel burnup because it is dominated by short-lived isotopes.

78. Fuel rod characteristics were for a Westinghouse 17 × 17–25 fuel assembly: uranium density, 9.25 g/cc; pellet radius, 0.41 cm; gap between fuel pellet and cladding, 0.008 cm; clad thickness, 0.057 cm; and outside radius of cladding, 0.475 cm (*Nuclear Fuel International*, Sept. 2001, pp. 24–25). Fuel composition as a function of burnup was calculated with ORIGEN 2.1. Criticality calculations were carried out with the MCNP4B2 code.

79. For 4.4 percent enriched fuel with a burnup of 13.25 MWd/kgHM, introduction of 1 one-cm of borated stainless steel (one percent boron by weight) between rows of fuel assemblies reduces the peak neutron multiplication factor $k_{eff}$ from 1.33 to 0.91. Fresh fuel would be barely critical ($k_{eff} = 1.05$) for a spacing of about 2 cm.

80. Criticality control with soluble boron creates the danger, however, of a criticality if a leaking pool is refilled with unborated water. Also, the water of BWRs must be free of boron. The pressure vessel and connected plumbing of a BWR would therefore have to be flushed after contact with boron-containing spent-fuel water.

81. *Spent Fuel Heatup Following Loss of Water During Storage*, p. 63.

82. *Ibid.*

83. *Op cit.*, p. 79.

84. A flow of 1 liter/sec can be maintained in a steel pipe with 2.5 cm inside diameter and a pressure drop of 0.015 atmosphere/m [*ASHRAE Handbook: Fundamentals* (American Society of Heating, Refrigeration and Air-conditioning Engineers, 2001), p. 35.6].

85. This may have been what a National Academy of Sciences committee had in mind when it stated "emergency cooling of the fuel in the case of attack could probably be accomplished using 'low tech' measures that could be implemented without significant

exposure of workers to radiation" [*Making the Nation Safer: The Role of Science and Technology in Countering Terrorism* (National Academy Press, 2002), p. 43]. One of our reviewers pointed out that a puncture hole in the stainless steel liner of the bottom of the Hatch nuclear power plant spent fuel pool caused by a dropped 350-pound core-shroud bolt in the mid 1990s was temporarily plugged with a rubber mat.

86. An interesting suggestion made by one of our reviewers also deserves further research: add to the escaping water a material such as is used to seal water-cooled automobile engines. Such sealant works by solidifying when it comes into contact with air.

87. The choice of age at transfer represents a tradeoff between cost and risk. We have picked five years based on the capabilities of existing dry storage systems.

88. The U.S. has approximately 100 GWe of nuclear capacity or about 1 GWe of capacity per spent-fuel pool. NAC projects that, in 2010, there will be 45,000 tons of spent fuel in pools (*US Spent Fuel Update: Year 2000 in Review* (Atlanta, Georgia: NAC Worldwide Consulting, 2001), i.e. an average of 450 tons per pool. In five years, a GWe of capacity discharges about 100 tons of fuel.

89. *2002 Summary of U.S. Generating Company In-pool Spent Fuel Storage Capability Projected Year that Full Core Discharge Capability Lost*," (Energy Resources International, 2002, www.nei.org/documents/Spent_Fuel_Storage_Status.pdf, accessed Dec. 14, 2002).

90. On average 350 tons of spent fuel would have to be removed from each of 100 pools (see note above). Spent fuel casks typically have a capacity of about 10 tons.

91. The dry storage casks currently licensed in the U.S. ⟨http://www.nrc.gov/reading-rm/doc-collections/cfr/part072/part072-0214.html⟩ are: **thick-walled:** General Nuclear Systems Castor V/21; **overpack:** Nuclear Assurance Corp. http://www.nacintl.com: NAC Storage/Transport (NAC S/T; NAC C-28 S/T); NAC Multipurpose Cannister System (NAC-MPS); NAC Universal Storage System (NAC-UMS); Transnuclear ⟨http://www.cogema-inc.com/subsidiaries/transnuclear.html⟩: NUHOMS horizontal modular storage system; Transnuclear TN-24, TN-32, and TN-68 Dry Storage Casks; Holtec http://www.holtecinternational.com: HI-STAR 100 and HI-STORM 100; British Nuclear Fuel Limited Spent Fuel Management System W-150 storage cask; and Pacific Sierra (now BNFL Fuel Solutions) Ventilated Storage Cask System VSC-24 ⟨http://www.bnfl.com⟩. See also *Information Handbook on Independent Spent Fuel Storage Installations* by M. G. Raddatz and M. D. Waters (Washington, DC: U.S. NRC, NUREG-1571, 1996).

92. F. Lange and G. Pretzsch, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH; E. Hoermann, Dornier GmbH; and W. Koch, Fraunhofer Institute for Toxicology and Aerosol Research, "Experiments to quantify potential releases and consequences from sabotage attack on spent fuel casks," 13th International Symposium on the Packaging and Transportation of Radioactive Material, Chicago Sept. 2001. Helium is often used to fill dry casks because of its superior heat-transfer characteristics and for leak detection. GNS-GNB did experiments in the 1980s to determine the temperature rise if helium leaked out of a Castor cask and was replaced by air. It was found that the maximum fuel rod temperature increased from about 400 to 460°C.

93. Helmut Hirsch and Wolfgang Neumann, "Verwundbarkeit von CASTOR-Behältern bei Transport und Lagerung," www.bund.net/lab/reddot2/pdf/studie_castorterror.rtf. (We are grateful to Hirsch for providing a summary in English.)

94. If the hole were not plugged, the $UO_2$ in the ruptured pins would begin to oxidize to $U_3O_8$, resulting in the pellets crumbling and releasing additional volatile fission products that could diffuse out of the hole ("History and actual status of aircraft impact and anti-tank weaponry consequences on spent fuel storage installations").

95. A ceramic "Ballistic Protection System" was tested successfully on a CASTOR cask by International Fuel Containers at the U.S. Army's Aberdeen Proving Grounds in June 1998 (Klaus Janberg, "History and actual status of aircraft impact and anti-tank weaponry consequences on spent fuel storage installations"). For a 100-ton cask, the shield would weigh at least 50 tons.

96. "History and actual status of aircraft impact and anti-tank weaponry consequences on spent fuel storage installations."

97. " the [6 cm] carbon steel liner 'balloons' and contracts the canister" ("Plane tough storage" by Michael McGough and Charles Pennington, *Nuclear Engineering International*, May 2002). The simulation assumes that the steel will stretch up to 37% at a stress of 30,000–70,000 psi (average of $3.4 \times 10^8$ pascals) without rupturing. The kinetic energy of a 400-kg shaft traveling at a speed of 220 m/sec is about $10^7$ joules. We have checked the plausibility of this result using a simplified geometry in which a flat circular sheet of steel 3.1 inches (8 cm) thick (taking into account the canister wall as well as the liner) and 1 meter in radius is stretched into a cone by keeping its edges fixed and pressing its center point in a direction perpendicular to the original plane of the sheet. In order for the sheet to absorb $10^7$ joules by stretching in this way, the center point would have to be pushed about 0.3 meters.

98. *Grenzüberschreitende UVP gemäß Art. 7 UVP-RL zum Standortzwischenlager Biblis; Bericht an das Österreichische Bundesministerium für Land- und Forstwirtschaft sowie an die Landesregierungen von Oberösterreich und Vorarlberg*, Federal Environment Agency, Vienna, Austria, February 2002; as well as corresponding reports by the Federal Environment Agency concerning the sites of Grafenrheinfeld, Gundremmingen, Isar, Neckar and Philippsburg. (We are grateful to H. Hirsch for providing us with an English summary of these reports.)

99. 3000 tons per year is the design capacity of the surface spent-fuel receiving facility at Yucca Mountain (Daniel Metlay, U.S. Nuclear Waste Technical Review Board, private communication, Nov 12, 2002). The rate of discharge of spent fuel from U.S. reactors is likely to decline only slowly during the next decades. Eight plants have already received 20-year license extensions from the NRC, 14 more have applications for extension under review, and, according the Nuclear Energy Institute, 26 more plan to apply for extensions by 2005, http://www.nei.org/doc.asp?catnum=3&catid=286.

100. The design capacity would be for 40,000 tons of spent fuel. The fuel handling capability would be about 200 casks or 2000 tonsU per year (Max De Long, Excel Energy, personal communication, November, 2002).

101. NAC estimates that the end-2000 US inventory of spent fuel was 42,900 tons, of which 2,430 tons was in dry storage. It estimates that the 2010 US inventory will be 64,300 tons, of which 19,450 tons will be in dry storage [*U.S. Spent Fuel Update: Year 2000 in Review* (Atlanta, Georgia: NAC Worldwide Consulting, 2001)]. The small increase in projected in-pool storage (4,400 tons) suggests that most U.S. spent-fuel pools are already approaching their dense-packed capacity.

102. We have assumed an average fuel burnup during 2005-10 of 43 MWd/kgU (the approximate average burnup in recent years), an average capacity factor of 0.85, and an

average heat to electrical power conversion efficiency of one third. With these assumptions, the amount of spent fuel discharged in 5 years is simply 100P metric tons, where P is the rated electrical generating capacity of the associated nuclear-power plant in GWe.

103. The cask is made out of ductile cast iron and has the following dimensions and weights: length, 5.45 m; outer diameter 2.44 m; cavity length, 4.55 m; cavity diameter, 1.48 m; wall thickness, 35 cm; empty weight, 104 tons; loaded weight 123 tons [*Transport and Storage Cask V/52* [GNS (Gesellschaft für Nuklear-Behälter mbH, 1997), p. 2, 4]. The CASTOR V/52 is similar to the CASTOR V/19 and V/21 except for being designed to accommodate internally 52 BWR fuel assemblies.

104. The metal canister in the NAC-UMS is made of stainless steel and can hold 24 PWR fuel assemblies or 56 BWR fuel assemblies. It is about 4.7 meters high, 1.7 meters in diameter, and has a wall thickness of 1.6 cm. The overpack is a reinforced-concrete cylinder about 5.5 meters high and 3.5 meters outside diameter. The wall of this overpack consists of a steel liner 6.4 cm thick and a layer of concrete 72 cm thick. Ambient air passes through vents in the overpack, and cools the outside of the metal container by natural convection.

105. NAC International could produce 180 casks per year within two-to-three years (Charles Pennington, NAC International, personal communication, November, 2002). Holtec could currently produce 200 casks per year and could increase this rate to about 300 casks per year (Chris Blessing, Holtec, private communication, November, 2002). We assume 10 tons average storage capacity per cask.

106. Based on discussions with cask manufacturers. The lower end of the range is for thin-walled casks with reinforced-concrete overpack. The upper end is for monolithic thick-walled casks equipped with missile shields.

107. Allison Macfarlane, "The problem of used nuclear fuel: Lessons for interim solutions from a comparative cost analysis," *Energy Policy, 29* (2001) pp. 1379–1389.

108. Assuming a burnup of 43 MWd/kgHM and a heat-to-electric-energy conversion ratio of one third.

109. *Monthly Energy Review, September 2002* [U.S. Department of Energy, Energy Information Administration, DOE/EIA-0035 (2002/09)], Table 9.9.

110. We thank one of our reviewers for pointing this out to us.

111. The walls and roof of the Gorleben building are about 50 and 15 cm thick reinforced concrete respectively (from Klaus Janberg).

112. NAC estimates that, by 2010, the U.S. will have 19,450 tons of spent fuel in dry storage (see note above). If we add 35,000 tons of older spent fuel from the storage pools, the total will be about 55,000 tons or about 550 tons per GWe of U.S. nuclear generating capacity.

113. The berms for the 300-cask site at the Palo Verde, Arizona nuclear power plant cost $5–10 million (Charles Pennington, NAC, private communication, November 2002).

114. With new NRC guidelines (ISG11, rev.2), which allow dry storage with peak cladding temperature up to 400°C, it is expected that a variant can be fielded with a capacity of 21 fuel assemblies with an average burnup of 60 MWd/tU (from Klaus Janberg).

115. In 2000, cask tests were being conducted with fuel burnups of up to 60 MWd/kgHM (Susan Shankman and Randy Hall, "Regulating Dry Cask Storage," *Radwaste Solutions*, July/August 2000, p. 10).

116. More than 25 nuclear power plants are today owned by such "limited-liability corporations" and additional corporate reorganizations are expected [*Financial Insecurity: The Increasing Use of Limited Liability Companies and Multi-Tiered Holding Companies to Own Nuclear Power Plants*, by David Schlissel, Paul Peterson and Bruce Biewald (Synapse Energy Economics, 2002), p. 1].

117. *Monthly Summary of Program Financial and Budget Information* (Office of Civilian Radioactive Waste Management, May 31, 2002). In 2001, U.S. nuclear power plants generated 769 million megawatt-hours net (*Monthly Energy Review, September 2002*, Table 8.1). With the enactment of the Gramm/Hollings/Rudman Budget Act in 1987, and the Budget Adjustment Act in 1990, the Nuclear Waste Fund ceased to be a standalone revolving fund. However, fees are placed in the General Fund Account of the U.S. Treasury and interest is accrued as if it were still a separate revolving account.

118. *Nuclear Waste Fund Fee Adequacy: An Assessment* (Department of Energy, DOE/RW- 0534, 2001). The report concludes that the revenues in the nuclear waste fund should be adequate but that there could be problems if interest rates fall significantly, or DOE incurs high settlement costs from lawsuits, or costs increase significantly.

119. The DOE negotiated with one utility company (PECO/Exelon) to take title to their spent fuel while it remained at the reactor and to pay for dry cask storage with money from the Nuclear Waste Fund. The US Court of Appeals for the 11th Circuit ruled, however, that DOE could not pay from the Fund to cover its own breach of its previous commitment under the Nuclear Waste Policy Act of 1982 to begin moving spent fuel from nuclear power plants to a deep underground repository by 1998 (Melita Marie Garza, 2002, "Exelon rivals win waste-suit round," *Chicago Tribune*, September 26, 2002 and Matthew Wald, 2002, "Taxpayers to owe billions for nuclear waste storage," *New York Times*, September 26, 2002.)

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# Comments on: "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States"

Allan S. Benjamin

I am one of the reviewers of the paper entitled: "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States," and am also the principal author of the Sandia report that is cited several times by the authors of the paper. The subject of spent-fuel pool vulnerabilities is a very important one in the present day environment, and I am pleased to be able to provide input. I think the paper correctly points out a problem that needs to be addressed, i.e., the fact that a loss of water from a high-density spent-fuel pool could have serious consequences. However, I also believe the paper falls short of addressing all the considerations that accompany the problem. Some of these considerations could affect the results of the cost-benefit analysis that is used to justify the authors' proposed solution: the re-racking of the pool to a low-density, open-lattice arrangement and the removal of the older fuel to dry storage casks. In a nutshell, the authors correctly identify a problem that needs to be addressed, but they do not adequately demonstrate that the proposed solution is cost-effective or that it is optimal.

On the plus side of the assessment, I agree with the authors' analysis of what would happen if there were a total loss of water from a high-density

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Dr. Benjamin was formerly a Distinguished Member of the Technical Staff at Sandia National Laboratories.

**Allan S. Benjamin**, Principal Scientist for Albuquerque Operations, ARES Corporation, 851 University Blvd. SE, Albuquerque, NM 87106. E-mail: abenjamin@arescorporation.com

spent-fuel pool that is packed wall-to-wall with zirconium-clad fuel. If some of that fuel had been recently discharged from a reactor core, there is not much doubt that the release of fission products to the environment would be significant. Our analyses in the referenced 1979 Sandia report did indeed show that the hottest part of the pool would heat up to the point where the cladding would first rupture and then ignite. Subsequent experiments we performed with electrically heated zirconium tubes (not formally reported) showed that there was a potential for a fire to propagate from hotter to colder fuel assemblies. It is not clear whether the fire would envelop the whole pool or just a part of it, but either way, the result would be undesirable.

I agree in principle with the calculations in the paper regarding the potential consequences of such an accident, except that it is unlikely that the whole inventory of fission products captured in the spent fuel would escape to the environment or that the wind would blow in one direction only (as assumed in the paper). Although there is clear evidence that some of the fuel would melt in such a situation, we don't know how much. Since we don't, it is conservative and appropriate to assume that a large fraction of the fission product inventory could become released to the environment. Whether that fraction is 0.20 or 1.00 doesn't change the fact that the release would be unacceptable.

It is also correct to say, as the authors have pointed out, that the situation could be even worse if enough water remained in the pool to cover the bottom of the storage racks so that air could not circulate, but not enough water to act as a significant heat sink for all of the decay heat produced by the fuel. This point was also made in the Sandia report.[1]

The authors' assessment of probabilities of occurrence is also reasonable in a bounding sense. They correctly point out that the likelihood of an accident leading to a critical loss of water is very low (estimated by the NRC to be less than one in 100,000 per pool per year). The probability of the same scenario resulting from a terrorist attack is unknown, and so the authors postulate a range of values. They point out, reasonably enough, that the upper end of the range could be significantly higher than the value for a loss of water initiated by an accident. I personally believe that the probability of a successful terrorist attack is very low, and I will give my reasons in a moment. Notwithstanding, the authors are correct in pointing out that the possibility of a terrorist attack is an issue that requires serious attention.

The problem occurs when the authors assert that these figures prove the cost effectiveness of their proposed solution. Before a judgment on cost effectiveness can be made, a variety of additional considerations have to be taken into account. These pervade all areas of the discussion: the calculation of the probabilities of occurrence, the resulting consequences, the effectiveness of the

proposed solution, the competing risks introduced by that solution, and the cost of implementation.

Let's talk first about the probability of a successful terrorist attack. The assumed situation is that the adversaries create a large hole in the spent-fuel pool, near the bottom of the pool, without dispersing the fuel or significantly deforming the racking structure. That situation is very unlikely. Using explosives or missiles, including the intentional crash of an aircraft, it would be difficult to accomplish a loss of almost all the water in the pool without disrupting the spent-fuel geometry. Significant damage to the racking structure or outright dispersal of the fuel would create a geometry that is more coolable by air flow and less susceptible to propagation of a zirconium fire than is the actual storage geometry.

Moreover, it would be very difficult for adversaries to achieve enough water loss by draining the pool even if they somehow gained direct access to the pool. The drain valves and gates are all located high enough to prevent the water from draining down to a dangerous level. As originally stated in the Sandia report and acknowledged in the paper, something like 75% of the height of the fuel rods would have to be uncovered for an overheating condition to result.

Gaining access to the pool in itself would be a very difficult proposition. The adversaries would have to figure out a way to avoid being detected by the on-site monitoring equipment and overcome by the on-site security forces. The probability of success in this venture can be analyzed using existing tools, but this has apparently not been done. Such tools exist at the company where I now work, ARES, and at the laboratory where I used to work, Sandia. Both have methods for identifying the pathways an adversary could take to a target and evaluating the probability of success associated with each pathway.

The upshot is that more work needs to be done in accounting for how an adversary's method of attack would change the initial conditions of the analysis, and in evaluating the adversary's likelihood of success.

Now let's discuss the consequences of a loss-of-water incident, which according to the paper could include "hundreds of billions of dollars" in property loss. An accurate accounting of costs versus benefits requires a best-estimate assessment of consequences, not a worst-case assessment. Normally, the evaluation is accomplished by formulating probability distributions to reflect the full range of radioactive releases that could emanate from the spent fuel pool and the full range of meteorological conditions that could affect the dispersion of that material. The most commonly-used result from this analysis is the mean consequence, which is obtained by sampling the probability distributions in a random fashion. It can reasonably be expected that the mean value of the expected property loss would be considerably lower than the worst-case value.

Let's now progress to the subject of evaluating the effectiveness of the proposed solutions. The main one given in the paper is to remove all the fuel that is more than five years old to dry storage casks and to re-rack the pool so that the remaining, younger spent fuel can be contained in a widely-spaced, open-lattice arrangement. The arguments in favor of that approach appear attractive. First, it assures that air cooling would be effective even if all the water were drained from the pool. Second, it reduces the inventory of the long-lived fission products remaining in the pool, so that even if all of them were dispersed to the environment, the long-term effects would be sharply reduced.

Several important factors are not considered here. First, as mentioned above, an adversary's attack involving an explosive, a missile, or an airplane crash that is serious enough to create a big hole in the spent-fuel pool would also probably disperse the fuel or at least rearrange the geometry. Therefore, the final configuration would not necessarily be more coolable than that for a high density pool subjected to the same insult. That leaves only the reduced fission product inventory as a definitive point of difference that could reduce the losses incurred from the event.

However, the results in the paper concerning radioactive contamination are flawed by the fact that the shorter-lived radioisotopes are not considered. Most notable among these are $^{131}$I, which has a half-life of 8 days, and $^{134}$Cs, which has a half-life of just over two years. Most of these radionuclides are contained within the younger fuel that still remains in the spent-fuel pool. While they do not contribute as highly to long-term property loss as the longer-lived isotope, $^{137}$Cs, they contribute more highly to early fatalities and latent cancer fatalities. Thus, a true cost-benefit accounting of the proposed solution must include consideration of these short-lived but very nasty radioisotopes.

Then there is the question of how effective the dry storage casks would be over a long period of time. The paper correctly acknowledges that an airplane crash into an array of dry storage casks could cause a release of radionuclides to the environment. It also presumes that only a few of the many casks in the array would be affected by the crash. Given the robust design of these casks, these observations are probably correct. However, the paper has failed to consider that many materials degrade or become brittle after a long exposure to radioactivity. Degradation or embrittlement can lead to leakage. Cask leakage has been a problem for some dry storage casks in the past, and the paper should acknowledge this. In performing a cost-benefit analysis, the risk from high probability, low consequence incidents, such as cask leakage, has to be considered along with the risk from low probability, high consequence incidents.

Finally, one must consider the competing risks. The process of removing such a large amount of fuel from the spent-fuel pool and transferring it to the

dry storage casks carries its own set of hazards. During the transfer process, both the probability of an accident and the degree of exposure in the event of a potential terrorist attack are greater than before or after the transfer. The paper suggests that the transfer would take place over a ten-year period. Someone needs to look at the question of vulnerability during that period.

Another competing risk can be identified for the authors' proposed design change, based on an earlier recommendation made in the Sandia report, to install emergency water sprays. The authors suggest that the hottest fuel should be stored along the sides of the pool, where the spray would be heaviest even if the building collapses on top of the pool. This argument ignores the fact that heat removal by air cooling is most effective when the hottest fuel is stored in the middle of the pool and the coolest fuel is stored along the sides. That arrangement promotes natural convective air flow currents, whereas the one being proposed in the paper inhibits them.

The question of implementation costs is one that I am not prepared to address at the present time. I would note, however, that special consideration needs to be given to the question of whether, on the basis of available space and security requirements, on-site dry storage of so much fuel is feasible at all reactor sites.

As a final but pivotal point, the evaluation of costs versus benefits should consider all plausible alternative risk reduction options. Certainly one such option is to accelerate the transfer of the spent fuel from spent-fuel pools directly to a permanent underground storage site. The paper claims that this process could take decades, given the controversial status of the Yucca Mountain project and the current budgetary limitations. However, if there is a national security issue at stake, Government projects can be accelerated. The Manhattan Project is a good example. It may turn out that when all risks and costs are taken into account, a direct transfer to underground storage is more cost-effective than a temporary transfer to on-site storage casks and a re-racking of the spent-fuel pools.

In summary, the authors are to be commended for identifying a problem that needs to be addressed, and for scoping the boundaries of that problem. However, they fall short of demonstrating that their proposed solution is cost-effective or that it is optimal.

## NOTE AND REFERENCE

1. Although most of the references made in the paper to the Sandia report are accurate, in the version reviewed by me, the first paragraph in the Introduction made two incorrect attributions. First, the accident evaluated in the Sandia study was a sudden loss of all the water, not a "sudden loss of water cooling." Loss of the water cooling system would

not result in the consequences cited by the authors since the water would remain as a large heat sink. Second, the Sandia report did not state that the loss-of-water scenario would lead to "the airborne release of massive quantities of fission products." Although zircaloy burning and some fuel melting would certainly occur, the Sandia study stopped short of evaluating, either qualitatively or quantitatively, the amount of fission products that would be released. Both of these points have now been corrected in the final version of the article.

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# THE AUTHORS RESPOND TO ALLAN BENJAMIN'S COMMENTS

Robert Alvarez, Jan Beyea, Klaus Janberg, Jungmin Kang, Ed Lyman, Allison Macfarlane, Gordon Thompson, Frank N. von Hippel

As the multiple references to it in our article attest, we have learned a great deal from the pioneering work of Allan Benjamin *et al, Spent Fuel Heatup Following Loss of Water During Storage* (NUREG/CR-0649; SAND77-1371 R-3, 1979). Indeed, many of our conclusions and recommendations essentially echo those made in that report 24 years ago, but never implemented because the probability of an accidental loss of water was estimated to be too low to justify action.

Benjamin argues that we should have estimated the probability that sabotage or terrorist attack might cause a loss of water. Indeed, he seems to suggest that the probability can be calculated with some precision with methods that his company offers. While we believe that systematic analysis is useful in identifying vulnerabilities, we are skeptical about the predictive value of probabilistic calculations—especially for malevolent acts.

We respond more briefly to Benjamin's other comments below:

**Magnitude of the release of $^{137}$Cs.** We looked at 10 and 100 percent releases—not just 100%.

**Sensitivity to the constant-wind assumption.** An estimate of the sensitivity of the contamination area to wind wander can be obtained by varying the opening angle in the wedge model calculation. Increasing the opening angle from 0.11 to 1 radians, for example, results in the area contaminated above 100 Ci/km2 increasing by about 20% for the 100% release and decreasing by about a factor of 3 for the 10% release.

**Feasibility of totally draining the pool through valves and gates.** We make no claim that this is possible. Rather we cite NRC staff concerns that a number of pools could be drained below the top of the spent fuel. This would result in very high radiation levels in the spent-fuel-pool building. Pools should

therefore be equipped with sources of makeup water that can be turned on from a remote location.

**Probabilities that terrorist attacks would put dense-packed fuel into a more coolable configuration and open-racked fuel into a less coolable configuration.** Benjamin makes both assertions. The first is far from obvious. With regard to the second, we point out that the assumption that the geometry of the spent fuel is not changed is a limitation of our analysis—as it is of all other analyses of which we are aware. The NRC should commission studies of the implications for coolability of potential changes in geometry.

**Omission of 8-day halflife $^{131}$I and 2-year halflife $^{134}$Cs in the consequence calculations.** Shorter-lived isotopes such as $^{131}$I and one-year halflife $^{106}$Ru could make significant contributions to short-term doses downwind from a spent-fuel-pool fire. However, our analysis was limited to the long-term consequences of such an accident where, as the consequences of the Chernobyl accident demonstrate, 30-year halflife $^{137}$Cs is the principle concern because it can force the evacuation of huge areas for decades.

**Effectiveness of dry casks over the long term.** We propose on-site dry-cask storage for about 30 years of older spent fuel that would, according to current plans, remain in pools for that length of time. Spent-fuel casks have already been in use for about 20 years and there is no evidence that they cannot last decades longer without significant deterioration.

**Risks during spent-fuel transfer.** We urge in the paper that these risks be carefully examined and minimized before the transfer begins. However, the fuel will have to be moved sooner or later in any case.

**Availability of space for dry-cask storage.** Nuclear power plants are surrounded by exclusion areas that provide ample space for a few tens of additional casks.

**Acceleration of Yucca Mtn. Project.** It would probably be counterproductive at this stage to try to significantly accelerate the licensing process of the Yucca Mountain underground spent-fuel repository. It would be worth exploring whether the delivery rate for spent-fuel could be increased above the current design rate of 3000 tons per year. However, there are so many political uncertainties associated with the transport of spent fuel to Yucca Mountain and so many technical issues that still have to be decided in its design and licensing process that speculation about possible acceleration should not be used as an excuse to ignore the relatively straightforward interim on-site storage option recommended in our paper.

# EXHIBIT 10

# NSIR/DPR-ISG-02

## INTERIM STAFF GUIDANCE

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## EMERGENCY PLANNING EXEMPTION REQUESTS

## FOR DECOMMISSIONING NUCLEAR POWER PLANTS

# TABLE OF CONTENTS

## 1.0 PURPOSE

The purpose of this interim staff guidance (ISG) is to provide guidance to U.S. Nuclear Regulatory Commission (NRC) staff in processing exemptions from the emergency preparedness (EP) requirements for nuclear power reactors that are undergoing the process of decommissioning. Licensees must follow the process outlined in 10 CFR 50.12 when applying for exemptions from EP regulations. Attachment 1 of this ISG should be used by the staff for reviewing the adequacy of the defueled onsite emergency plan submitted by a licensee. The staff should use this ISG until it is superceded or incorporated into other guidance or rulemaking.

The NRC issues guidance to describe and make available to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations. The guidance is not a substitute for regulations, and compliance with it is not required. Methods that differ from those set forth in guidance may also be deemed acceptable if they conform to the regulations and provide the basis for licensing decisions.

## 2.0 SCOPE

This ISG reflects the changes made to sections 50.47(b) and 50.54(q) of Title 10 of the *Code of Federal Regulations* (10 CFR) and Appendix E to 10 CFR Part 50 issued on November 23, 2011 (76 *Federal Register* (FR) 72560). This guidance is only applicable to a nuclear power reactor that has notified the NRC that it has permanently ceased operation in accordance with 10 CFR 50.82(a)(1)(i), has certified permanent removal of fuel from the reactor vessel under 10 CFR 50.82(a)(1)(ii), is storing spent fuel in a spent fuel pool (SFP) and is not located on the site of an operating nuclear power reactor. The Office of Nuclear Materials Safeguards and Security Spent Fuel Project Office Interim Staff Guidance – 16, "Emergency Planning," provides the appropriate guidance for fuel stored in a dry cask storage facility.

## 3.0 BACKGROUND

The EP requirements in 10 CFR Part 50 that apply to licensees of operating nuclear power reactors also apply to decommissioning power reactor licensees because these licensees retain their part 50 operating licenses or part 52 combined licenses after permanent cessation of operations and removal of fuel from the reactor vessel. The staff recognizes that the risk of a large offsite radiological release at a decommissioning power reactor storing irradiated fuel in the SFP is lower than the risk of a large offsite radiological release from an operating power reactor and its SFP, based on the consideration of initiating reactor events associated with normal and abnormal operations, design-basis accidents, and certain beyond design-basis accidents applicable to a decommissioning site. For example, in NUREG-1738, "The Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," the NRC determined for spent fuel aged one year, a risk factor of a zirconium fire initiated by a seismic event at $2 \times 10^{-7}$ to $2 \times 10^{-6}$ for the plants studied. In contrast, at operating reactors additional risk-significant accidents for which EP is expected to provide dose savings are on the order of $1 \times 10^{-5}$ per year. Because of the lower comparative risk from a decommissioning power reactor, licensees typically make a case for an exemption on the basis that the application of the regulation in the particular circumstance decommissioning plants is not necessary to achieve the underlying purpose of the rule.

In the 1990s, the staff developed a thermal-hydraulic criterion for determining when reductions in EP requirements at decommissioning plants could be permitted. The criterion was used on a

case-by-case basis to grant exemptions from certain EP requirements. The criterion was based on demonstrating that spent fuel stored in the SFP would sufficiently air-cool and would not reach the zirconium ignition temperature if the water in the pool were to be fully drained or there was at least ten hours to take action to recover SFP inventory and take ad hoc actions to protect the public. NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82", and NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR [boiling water reactor] and PWR [pressurized water reactor] Permanently Shutdown Nuclear Power Plants", provides temperatures associated with the self-initiation and propagation of zirconium fires.

In SECY-97-120, "Rulemaking Plan for Emergency Planning Requirements for Permanently Shutdown Nuclear Power Plant Sites 10 CFR Part 50.54(q) and (t); 10 CFR 50.47; and Appendix E to 10 CFR Part 50," the staff presented the Commission with a rulemaking plan to amend the EP requirements for permanently shutdown nuclear power plant (NPP) sites. SECY-00-0145, "Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning," subsequently included sample rule language for EP at decommissioning plants. Because of the uncertainties associated with the risk and time frame for zirconium fire vulnerability as stated in SECY-00-0145, the staff suspended its decommissioning rulemaking efforts until the associated technical issues could be satisfactorily resolved.

In January 2001, the NRC published NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," providing a technical basis for the decommissioning rulemaking for permanently shutdown nuclear power plants proposed in SECY-00-0145. NUREG-1738 contained the results of the staff's evaluation of the potential accident risk for a SFP at a decommissioning power reactor in the United States. Specifically, NUREG-1738 stated that fuel assembly geometry and rack configuration are plant specific, and both are subject to unpredictable changes after an earthquake or cask drop that drains the pool. Therefore, because a non-negligible decay heat source lasts many years and configurations ensuring sufficient air flow for cooling cannot be assured, the possibility of reaching the zirconium ignition temperature cannot be precluded on a generic basis.

In SECY-01-0100, "Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools," the staff concluded that there was no immediate safety concern or need for immediate regulatory action for existing decommissioning power reactor licensees that had been previously granted EP exemptions. These conclusions were based on a review of the site-specific conditions at each existing decommissioning plant's power reactor and the low probability of the beyond-design-basis conditions occurring that would be necessary to initiate a zirconium fire.

In a memorandum dated August 16, 2002, the staff notified the Commission that it had discontinued the integrated rulemaking for decommissioning power reactors and generic regulatory activities because of the apparent lack of future licensees that would benefit from such regulations at that time and the need to devote resources to security related issues due to the events of September 11, 2001. Additionally, the staff provided that if any operating power reactors were to shutdown permanently, decommissioning regulatory issues would continue to be addressed on an ad hoc basis through the exemption process in a manner based on the then-current practice.

Attachment 2 provides a listing of decommissioning power reactors and bases provided in support of reducing EP requirements, specifically the elimination of formal offsite EP requirements.

## 4.0 OVERVIEW OF EXISTING GUIDANCE

The NRC published NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," in August 1997, providing recommendations on operationally-based regulations that could be partially or totally removed for decommissioning power reactor licensees without impacting public health and safety. It recommended that licensees apply for exemptions from the following offsite emergency planning requirements, after the fuel is no longer susceptible to substantial zircaloy oxidation and the fuel cladding will remain intact given the SFP is drained:

- The early public notification requirements (§50.47(b)(5) and Appendix E, section IV.D.3);
- The periodic dissemination of emergency planning information to the public (§50.47(b)(7) and Appendix E, section IV.E.8);
- Offsite emergency facilities and equipment such as the EOF, and the emergency news center (§50.47(b)(8), Appendix E, section IV.E.8);
- Offsite radiological assessment and monitoring capability, including field teams (§50.47(b)(9));
- Periodic offsite drills and exercises (§50.47(b)(14), Appendix E, section IV.F.3); and
- Licensee headquarters support personnel training (§50.47(b)(15), Appendix E, section IV.F.b.h).

NUREG-1738 identified a zirconium fire resulting from a substantial loss of water from the SFP as the only postulated scenario at a decommissioning plant that could result in a significant release. The scenarios that lead to this condition have very low probabilities of occurrence and are considered beyond design-basis accidents; however, the consequences of such accidents could lead to an offsite dose in excess of the U.S. Environmental Protection Agency's (EPA) protective action guidelines (PAGs). The risk associated with zirconium fire events decreases as decay time increases and decay heat decreases. In SECY-01-0100, the staff proposed maintaining a level of offsite EP consistent with the Commission's defense-in-depth philosophy while utilizing the risk insights of NUREG-1738.

As the spent fuel ages, the generation of decay heat decreases. After a certain amount of time, the overall risk of a zirconium fire becomes insignificant due to two factors: 1) the amount of time available for preventative and mitigating actions, and, 2) the increased probability that the fuel is air coolable. This lower risk supports the reduction of EP requirements as described in Table 1.

In SECY-01-0100, the staff proposed regulations for maintaining a level of offsite EP consistent with the Commission's defense-in-depth philosophy while utilizing the risk insights of NUREG-1738. 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 PWR fuel has decayed at least 60 days.

In June 2013, a draft study, entitled "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boiling Water Reactor," was published for public

comment. The purpose of the consequence study was to determine if accelerated transfer of older, colder spent fuel from the SFP at a reference plant to dry cask storage significantly reduces risks to public health and safety. The specific reference plant used for the study was a General Electric Type 4 BWR with a Mark I containment.

The study states: “Past risk studies have shown that storage of spent fuel in a high-density configuration is safe and risk of a large release due to an accident is very low. This study’s results are consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking. The NRC continues to believe, based on this study and previous studies that spent fuel pools protect public health and safety.”

The study also estimated that the likelihood of a radiological release from the SFP resulting from the selected severe seismic event analyzed in the study was on the order of one time in 10 million years or lower. The study analyzed two cases for each scenario: one where mitigation measures of 10 CFR 50.54(hh)(2) were credited, and one where they were not used or were unsuccessful. It showed that successful mitigation reduces the likelihood of a release and that the likelihood of a release was equally low for both high- and low-density loading in the SFP. The study did not consider the post-Fukushima mitigation measures required by Orders EA-12-049 (Mitigating Strategies Order) and EA-12-051 (Reliable Hardened Containment Vents Order)

Additionally, the NRC conducted research to assess the risk to the public and identify the dominant contributors to that risk for moving spent fuel to dry cask storage. NUREG-1864, “A Pilot Probabilistic Risk Assessment [PRA] of a Dry Cask Storage System at a Nuclear Power Plant,” was published in March 2007. The staff analyzed risk by selecting a specific cask system at a specific BWR site, developed a comprehensive list of initiating events, and evaluated the risk associated with each initiating event. Initiating events considered included the dropping of the cask inside the secondary containment building during transfer operations, as well as external events during onsite storage (such as earthquakes, floods, high winds, lightning strikes, accidental aircraft crashes, and pipeline explosions). Potential cask failures from mechanical and thermal loads, including thermal loads caused by mis-loading events, were also modeled. In the event of a cask failure/breach, the fuel inventory available for release was based on 10 year old fuel. Weather conditions and the population distribution in the vicinity of the selected site were also considered.

The results of PRA studies are normally presented in measures such as the probability of a prompt fatality and the probability of a latent cancer fatality. The results of this study indicated that no prompt fatalities would be expected. The resulting calculated risk for a latent cancer fatality was extremely small (i.e., less than one in a trillion years). Due to the exceedingly low risk numbers calculated, the conclusion that should be reached is that cask storage systems provide a safe means to store spent nuclear fuel**.**

## 5.0 EVALUATION OF EXEMPTIONS TO EP REGULATIONS

Consistent with previous exemption requests informed by the most recent SFP studies, the NRC should not grant approval for the exemption of EP requirements for decommissioning power reactor licensees until site-specific analyses provide sufficient assurance that an offsite radiological release is not postulated to exceed the EPA PAGs at the site boundary, or that there is sufficient time to initiate appropriate mitigating actions by offsite agencies on an ad hoc basis to protect the health and safety of the public. The expected analysis will include the

amount of time that lapses from when the SFP drains and air flow passages are blocked to when the hottest fuel assembly reaches 900 degrees Celsius. The staff concluded in SECY-00-0145 that, because of the considerable time available to initiate and implement mitigative actions, or if necessary, protective actions, formal emergency plans for rapid initiation and implementation of protective actions are no longer needed. For SFPs, after one year of decay time, in the case of an event that could lead to a zirconium fire, licensees would have 10 to 12 hours, which can be considered by NRC staff to be a sufficient amount of time to implement appropriate mitigative measures, as well as, offsite protective actions, if necessary, without preplanning.

In addition to the SFP analysis, any accident analyses in the FSAR that is still applicable in the defueled condition of the plant, such as a fuel handling accident, should be reviewed and any accidents no longer bounded by previous analyses should be analyzed. Historically, exemption requests have included analyses of expended resin fires and direct radiation exposure due to a drained SFP.

The analyses and conclusions described in NUREG-1738 are predicated on the risk reduction measures identified in the study as Industry Decommissioning Commitments (IDC) and Staff Decommissioning Assumptions (SDA), listed in Attachment 2. The staff should ensure that the licensee has addressed these IDCs and SDAs in the final safety analysis report for the decommissioning site if they are storing fuel in a SFP. The staff should verify the licensee presents a determination that there is sufficient time, resources and personnel available to initiate mitigative actions that will prevent an offsite release that exceeds EPA PAGs. The determination must also include a spent fuel heat up analysis for a loss of inventory event leading to fuel uncoverery with obstructed air flow (adiabatic heat-up).

Table 1 depicts the potential exemption requests, based on the staff's experience, for the time period beginning approximately 12 months after the final reactor shutdown, when the only event that could lead to an offsite dose exceeding EPA PAGs is a zirconium fire and the licensee has sufficient time to initiate mitigating actions for the event. The licensee must provide an analysis which indicates that fuel in the SFP meets these conditions. Differences or deviations from Table 1, "Exemptions for Consideration," will be reviewed on a case-by-case basis.

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

| ~~Strikethrough~~ text indicates requested exemptions to rule language. | |
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| **10 CFR 50.47 Emergency Plans** | **Basis for Change** |
| (b) The onsite ~~and, except as provided in paragraph (d) of this section, offsite~~ emergency response plans for nuclear power reactors must meet the following standards: | In the Statement of Considerations for the Final Rule for EP requirements for ISFSIs and for MRS facilities (60 FR 32430; June 22, 1995), the Commission responded to comments concerning offsite emergency planning for ISFSIs or an MRS and concluded that, "the offsite consequences of potential accidents at an ISFSI or a MRS [monitor retrievable storage installation] would not warrant establishing Emergency Planning Zones." In a nuclear power reactor's permanently defueled state, the accident risks are more similar to an ISFSI or MRS than an operating nuclear power plant. The draft proposed rulemaking in SECY-00-0145 suggested that after at least one year of spent fuel decay time, the decommissioning licensee would be able to reduce its EP program to one similar to that required for an MRS under 10 CFR 72.32(b) and additional EP reductions would occur when: (1) approximately five years of spent fuel decay time has elapsed; or (2) a licensee has demonstrated that the decay heat level of spent fuel in the pool is low enough that the fuel would not be susceptible to a zirconium fire for all spent fuel configurations. The EP program would be similar to that required for an ISFSI under 10 CFR 72.32(a) when fuel stored in the SFP has more than five years of decay time and would not change substantially when all the fuel is transferred from the SFP to an onsite ISFSI. Exemptions from offsite EP requirements have been approved when the specific site analyses show that at least ten hours is available from a partial drain down event where cooling of the spent fuel is not effective until the hottest fuel assembly reaches 900°C. Because ten hours allows sufficient time to initiate mitigative actions to prevent a zirconium fire in the SFP or to initiate ad hoc offsite protective actions, offsite EP plans are not necessary for these permanently defueled nuclear power plant licensees. |
| (1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations ~~within the Emergency Planning Zones~~ have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis. | See basis for 50.47(b). |
| (3) Arrangements for requesting and effectively using assistance resources have | Decommissioning power reactors present a low likelihood of any credible accident resulting in |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| been made, ~~arrangements to accommodate State and local staff at the licensee's Emergency Operations Facility have been made,~~ and other organizations capable of augmenting the planned response have been identified. | radiological releases requiring offsite protective measures because of the permanently shut down and defueled status of the reactor. An emergency operations facility would not be required. The "nuclear island" or "control room" or other location can provide for the communication and coordination with offsite organizations for the level of support required.<br><br>Also see basis for 50.47(b). |
| (4) A standard emergency classification and action level scheme, the basis of which include facility system and effluent parameters, is in use by the nuclear facility licensee~~, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures~~. | EALs are to be consistent with Section 8 (if applicable) and Appendix C of NEI 99-01 Revision 6 endorsed by the NRC in a letter dated March 28, 2013. No offsite protective actions are anticipated to be necessary, so classification above the Alert level is no longer required.<br><br>Also see basis for 50.47(b). |
| (5) Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all organizations; the content of initial and follow up messages to response organizations ~~and the public~~ has been established~~; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established~~. | Per SECY-00-0145, after approximately 1 year of spent fuel decay time [and as supported by the licensee's SFP analysis], the staff believes an exception to the offsite EPA PAG standard is justified for a zirconium fire scenario considering the low likelihood of this event together with time available to take mitigative or protective actions between the initiating event and before the onset of a postulated fire. The spent fuel scoping study provides that depending on the size of the pool liner leak, releases could start anywhere from eight hours to several days after the leak starts, assuming that mitigation measures are unsuccessful. If 10 CFR 50.54(hh)(2) type of mitigation measures are successful, releases could only occur during the first several days after the fuel came out of the reactor. Therefore, offsite EP plans are not necessary for these permanently defueled nuclear power plant licensees.<br><br>Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor, June, 2013 |
| (6) Provisions exist for prompt communications among principal response organizations to emergency personnel ~~and to the public~~. | See basis for 50.47(b). |
| (7) ~~Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors),~~ [T]he principal points of contact with the news media for dissemination of information during an | See basis for 50.47(b). |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| emergency ~~(including the physical location or locations)~~ are established in advance, and procedures for coordinated dissemination of information to the public are established. | |
| (9) Adequate methods, systems, and equipment for assessing and monitoring actual or potential ~~offsite~~ consequences of a radiological emergency condition are in use. | See basis for 50.47(b) |
| (10)~~ A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. In developing this range of actions, consideration has been given to evacuation, sheltering, and, as a supplement to these, the prophylactic use of potassium iodide (KI), as appropriate. Evacuation time estimates have been developed by applicants and licensees. Licensees shall update the evacuation time estimates on a periodic basis. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.~~ | In the unlikely event of a SFP accident, the iodine isotopes which contribute to an off-site dose from an operating reactor accident are not present, so potassium iodide (KI) distribution off-site would no longer serve as an effective or necessary supplemental protective action.<br>The Commission responded to comments in its Statement of Considerations for the Final Rule for emergency planning requirements for ISFSIs and MRS facilities (60 FR 32435), and concluded that, “the offsite consequences of potential accidents at an ISFSI or a MRS would not warrant establishing Emergency Planning Zones.” Additionally, in the Statement of Considerations for the Final Rule for EP requirements for ISFSIs and for MRS facilities (60 FR 32430), the Commission responded to comments concerning site-specific emergency planning that includes evacuation of surrounding population for an ISFSI not at a reactor site, and concluded that, “The Commission does not agree that as a general matter emergency plans for an ISFSI must include evacuation planning.”<br><br>Also see basis for 50.47(b). |
| (c)(2) ~~Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.~~ The size of the EPZs ~~also~~ may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. ~~The plans for the ingestion pathway shall focus on such actions as are appropriate to protect the food ingestion pathway.~~ | See basis for 50.47(b). |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| **10 CFR Part 50, Appendix E, section IV** | **Basis for Change** |
| 1. The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, i.e., organization for coping with radiological emergencies, assessment actions, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, and recovery~~, and onsite protective actions during hostile action~~. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license under this Part, or for an early site permit (as applicable) or combined license under 10 CFR Part 52, shall contain information needed to demonstrate compliance with the standards described in § 50.47(b), and they will be evaluated against those standards. | The EP Final Rule published in the Federal Register (76 FR 72560; November 23, 2011) amended certain requirements in 10 CFR Part 50. Among the changes, the definition of “hostile action” was added as an act directed toward an NPP or its personnel. This definition is based on the definition of "hostile action" provided in NRC Bulletin 2005-02. NRC Bulletin 2005-02 was not applicable to nuclear power reactors that have permanently ceased operations and have certified that fuel has been removed from the reactor vessel.<br><br>The NRC excluded non-power reactors (NPR) from the definition of "hostile action" at that time because an NPR is not a nuclear power plant and a regulatory basis had not been developed to support the inclusion of non-power reactors in that definition. Likewise, an SFP and an ISFSI are not nuclear power plants as defined in the NRC’s regulations. The staff also considered the similarities between a decommissioning NPP and a non-power reactor to determine whether they should be included within the definition of “hostile action.” NPRs pose lower radiological risks to the public from accidents than do power reactors because: (1) the core radionuclide inventories are lower as a result of their lower power levels and often shorter operating cycle lengths; and (2) NPRs have lower decay heat associated with a lower risk of core melt and fission product release in a loss-of-coolant accident. A decommissioning power reactor also has a low likelihood of a credible accident resulting in radiological releases requiring offsite protective measures. For all of these reasons, the staff concludes that a decommissioning power reactor is not a facility that falls within the definition of “hostile action.” |
| 2.~~ This nuclear power reactor license applicant shall also provide an analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, using the most recent U.S. Census Bureau data as of the date the applicant submits its application to the NRC.~~ | See basis for 50.47(b)(10). |
| 3.~~ Nuclear power reactor licensees shall use NRC approved evacuation time estimates (ETEs) and updates to the ETEs in the formulation of protective action recommendations and shall provide the ETEs and ETE updates to State and local~~ | See basis for IV.2. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| ~~governmental authorities for use in developing offsite protective action strategies.~~ | |
| 4.~~ Within 365 days of the later of the date of the availability of the most recent decennial census data from the U.S. Census Bureau or December 23, 2011, nuclear power reactor licensees shall develop an ETE analysis using this decennial data and submit it under § 50.4 to the NRC. These licensees shall submit this ETE analysis to the NRC at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies~~ | See basis for IV.2. |
| 5.~~ During the years between decennial censuses, nuclear power reactor licensees shall estimate EPZ permanent resident population changes once a year, but no later than 365 days from the date of the previous estimate, using the most recent U.S. Census Bureau annual resident population estimate and State/local government population data, if available. These licensees shall maintain these estimates so that they are available for NRC inspection during the period between decennial censuses and shall submit these estimates to the NRC with any updated ETE analysis.~~ | See basis for IV.2. |
| 6.~~ If at any time during the decennial period, the EPZ permanent resident population increases such that it causes the longest ETE value for the 2-mile zone or 5-mile zone, including all affected Emergency Response Planning Areas, or for the entire 10-mile EPZ to increase by 25 percent or 30 minutes, whichever is less, from the nuclear power reactor licensee's currently NRC approved or updated ETE, the licensee shall update the ETE analysis to reflect the impact of that population increase. The licensee shall submit the updated ETE analysis to the NRC under § 50.4 no later than 365 days after the licensee's determination that the criteria for updating the ETE have been met and at least 180 days before using it to form protective action recommendations and providing it to State and local governmental authorities for use in developing offsite protective action strategies.~~ | See basis for IV.2. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| **10 CFR Part 50, Appendix E, section IV.A** | **Basis for Change** |
| A.1. A description of the normal plant ~~operating~~ organization. | Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," states in part: "… there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified." In Appendix A, a nuclear power unit is defined as a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Based on the permanently shut down and defueled status of the reactor, a decommissioning reactor is not a facility that can be operated to generate electrical power. Therefore, it does not have a "plant operating organization." |
| A.3. ~~A description, by position and function to be performed, of the licensee's headquarters personnel who will be sent to the plant site to augment the onsite emergency organization.~~ | The number of staff at decommissioning sites is generally small but is commensurate with the need to safely store spent fuel at the facility in a manner that is protective of public health and safety. Decommissioning sites typically have a level of emergency response that does not require response by headquarters personnel. |
| A. 4. Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making ~~offsite~~ dose projections, and a description of how these projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental entities. | Although, the likelihood of events that would result in doses in excess of the EPA PAGs to the public beyond the owner controlled area boundary based on the permanently shut down and defueled status of the reactor is extremely low, the licensee still must be able to determine if a radiological release is occurring. If a release is occurring, then the licensee staff should promptly communicate that information to offsite authorities for their consideration. The offsite organizations are responsible for deciding what, if any, protective actions should be taken. |
| A. 5. ~~Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications, such as consultants, who are not employees of the licensee and who may be called upon for assistance for emergencies shall also be identified. The special qualifications of these persons shall be described.~~ | The number of staff at decommissioning sites is generally small but should be commensurate with the need to operate the facility in a manner that is protective of public health and safety. |
| A.7. ~~By June 23, 2014,~~ identification of~~, and a description of the~~ assistance expected from, appropriate State, local, and Federal agencies | Requiring a licensee for a decommissioning site to provide a description of the assistance expected from appropriate State, local, and Federal agencies with |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| with responsibilities for coping with emergencies, ~~including hostile action at the site. For purposes of this appendix, "hostile action" is defined as an act directed toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force.~~ | responsibilities for coping with emergencies is an unnecessary burden on the licensee, in light of the low risk of an emergency necessitating offsite assistance.<br><br>Requiring a licensee to identify and describe the assistance expected from appropriate State, local, and Federal agencies with responsibilities for coping with hostile action at the site is unnecessary because, as explained in section IV.1, a decommissioning power reactor licensee is exempt from requirements in Appendix E related to a "hostile action." |
| A.8. ~~Identification of the State and/or local officials responsible for planning for, ordering and controlling appropriate protective actions, including evacuations when necessary.~~ | Offsite emergency measures are limited to support provided by local police, fire departments, and ambulance and hospital services as appropriate. Since EPA PAGs are not expected to be exceeded offsite, protective actions such as evacuation should not be required.<br><br>Also see basis for 50.47(b)(10) |
| A.9. ~~By December 24, 2012, for nuclear power reactor licensees, a detailed analysis demonstrating that on-shift personnel assigned emergency plan implementation functions are not assigned responsibilities that would prevent the timely performance of their assigned functions as specified in the emergency plan.~~ | The number of staff at decommissioning sites is generally small but should be commensurate with the need to operate the facility in a manner that is protective of public health and safety. Responsibilities should be well defined in the emergency plan and procedures, regularly tested through drills and exercises audited and inspected by the licensee and the NRC. The duties of the onshift personnel at a decommissioning reactor facility are not as complicated and diverse as those for an operating reactor.<br><br>The staff considered the similarity between the staffing levels at a permanently shutdown and defueled reactor and staffing levels at NPRs. The minimal systems and equipment needed to maintain the spent nuclear fuel in the spent fuel pool or in a dry cask storage system in a safe condition requires minimal personnel and is governed by Technical Specifications. In the EP Final Rule, the NRC agreed that the staffing analysis requirement was not necessary for non-power reactor licensees due to the small staffing levels required to operate the facility. For all of these reasons, the staff concludes that a decommissioning NPP is exempt from the requirement of 10 CFR Part 50, Appendix E, Section IV.A.9. |
| **10 CFR Part 50, Appendix E, section IV.B** | **Basis for Change** |
| 1. The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as | EALs are to be consistent with Appendix 1 (if applicable) and Appendix C of NEI 99-01, Revision 6, "Methodology for Development of Emergency Action Levels."<br><br>Also see basis for section IV.1. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within ~~and outside~~ the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite ~~and offsite~~ monitoring. ~~By June 20, 2012, for nuclear power reactor licensees, these action levels must include hostile action that may adversely affect the nuclear power plant~~. The initial emergency action levels shall be discussed and agreed on by the applicant or licensee and State and local governmental authorities, and approved by the NRC. Thereafter, emergency action levels shall be reviewed with the State and local governmental authorities on an annual basis. | |
| 10 CFR Part 50, Appendix E, section IV.C | |
| 1. The entire spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite ~~and offsite~~ radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, ~~such as the pressure in containment and the response of the Emergency Core Cooling System~~) for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) notification of unusual events, (2) alert, ~~(3) site area emergency, and (4) general emergency~~ of 10 CFR Part 50, Appendix E, IV.C.1. These classes are further discussed in NUREG-0654/FEMA-REP-1. | Containment parameters do not provide an indication of the conditions at a defueled facility and emergency core cooling systems are no longer required. Other indications such as SFP level or temperature can be used at sites where there is spent fuel in the SFPs.<br><br>In the Statement of Considerations for the Final Rule for EP requirements for ISFSIs and for MRS facilities (60 FR 32430), the Commission responded to comments concerning a general emergency at an ISFSI and MRS, and concluded that, "…an essential element of a General Emergency is that a release can be reasonably expected to exceed EPA Protective Action Guidelines exposure levels off site for more than the immediate site area." The probability of a condition reaching the level above an emergency classification of alert is very low. In the event of an accident at a defueled facility that meets the conditions for relaxation of EP requirements, there will be time to take ad hoc measures to protect the public."<br><br>As stated in NUREG-1738, for instances of small SFP leaks or loss of cooling scenarios, these events evolve very slowly and generally leave many days for recovery efforts. Offsite radiation monitoring will be performed as the need arises. Due to the decreased risks associated with defueled plants, offsite radiation monitoring systems are not required. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| | EALs should to be developed with the guidance provided in NEI 99-01, Revision 6. |
| 2.~~ By June 20, 2012, nuclear power reactor~~ licensees shall establish and maintain the capability to assess, classify, and declare an emergency condition~~ within 15 minutes~~ after the availability of indications to plant operators that an emergency action level has been exceeded and shall promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. Licensees shall not construe these criteria as a grace period to attempt to restore plant conditions to avoid declaring an emergency action due to an emergency action level that has been exceeded. Licensees shall not construe these criteria as preventing implementation of response actions deemed by the licensee to be necessary~~ to protect public health and safety provided that any delay in declaration does not deny the State and local authorities the opportunity to implement measures necessary to protect the public health and safety.~~ | In the Proposed Rule (74 FR 23254) to amend certain emergency planning requirements for 10 CFR Part 50, the NRC asked for public comment on whether the NRC should add requirements for non-power reactor licensees to assess, classify, and declare an emergency condition within 15 minutes and promptly declare an emergency condition. The NRC received several comments on these issues. The NRC believes there may be a need for the NRC to be aware of security related events early on so that an assessment can be made to consider the likelihood that the event is part of a larger coordinated attack. However, the NRC determined that further analysis and stakeholder interactions are needed prior to changing the requirements for non-power reactor licensees. Therefore, the NRC did not include requirements in the 2011 EP Final Rule for non-power reactor licensees to assess, classify, and declare an emergency condition within 15 minutes and promptly declare an emergency condition. The staff considered the similarity between a permanently defueled reactor and a non-power reactor for the low likelihood of any credible accident resulting in radiological releases requiring offsite protective measures. |
| **10 CFR Part 50, Appendix E, section IV.D** | **Basis for Change** |
| 1. Administrative and physical means for notifying local, State, and Federal officials and agencies ~~and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary,~~ shall be described. This description shall include identification ~~of the appropriate officials, by title and agency,~~ of the State and local government agencies ~~within the EPZs.~~ | See basis for 50.47(b) and 50.47(b)(10). |
| 2.~~ Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall~~ | See basis for section IV.D.1. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| ~~also be used to disseminate to any transient population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.~~ | |
| 3. A licensee shall have the capability to notify responsible State and local governmental agencies within ~~15~~ minutes after declaring an emergency. ~~The licensee shall demonstrate that the appropriate governmental authorities have the capability to make a public alerting and notification decision promptly on being informed by the licensee of an emergency condition. Prior to initial operation greater than 5 percent of rated thermal power of the first reactor at the site, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public with the plume exposure pathway EPZ. The design objective of the prompt public alert and notification system shall be to have the capability to essentially complete the initial alerting and notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this alerting and notification capability will range from immediate alerting and notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the appropriate governmental authorities to make a judgment whether or not to activate the public alert and notification system. The alerting and notification capability shall additionally include administrative and physical means for a backup method of public alerting and notification capable of being used in the event the primary method of alerting and notification is unavailable during an emergency to alert or notify all or portions of the plume exposure pathway EPZ population. The backup method shall have the capability to alert and notify the public within the plume exposure pathway EPZ, but does not need to meet the 15 minute design objective for the primary prompt public alert and notification system. When there is a decision to activate the alert and notification system, the appropriate governmental authorities will~~ | While the capability needs to exist for the notification of offsite government agencies within a specified time period, previous exemptions have allowed for extending the State and local government agencies' notification time up to 60 minutes based on the site-specific justification provided. A specific notification time should be provided and justified, as part of the exemption request.<br><br>Also see basis for 50.47(b) and 50.47(b)(10). |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| ~~determine whether to activate the entire alert and notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public alert and notification system shall remain with the appropriate governmental authorities.~~ | |
| 4. ~~If FEMA has approved a nuclear power reactor site's alert and notification design report, including the backup alert and notification capability, as of December 23, 2011, then the backup alert and notification capability requirements in Section IV.D.3 must be implemented by December 24, 2012. If the alert and notification design report does not include a backup alert and notification capability or needs revision to ensure adequate backup alert and notification capability, then a revision of the alert and notification design report must be submitted to FEMA for review by June 24, 2013, and the FEMA-approved backup alert and notification means must be implemented within 365 days after FEMA approval. However, the total time period to implement a FEMA-approved backup alert and notification means must not exceed June 22, 2015.~~ | See basis for section IV D.3. regarding the alert and notification system requirements. |
| **10 CFR Part 50, Appendix E, section IV.E** | **Basis for Change** |
| 8.a.(i) A licensee ~~onsite technical support center and an emergency operations~~ facility from which effective direction can be given and effective control can be exercised during an emergency; | Due to the low probability of design-basis accidents or other credible events to exceed the EPA PAGs, the significantly reduced staff and the minimal expected offsite response required, offsite agency response will not be required at an emergency operations facility (EOF) and onsite actions may be directed from the control room or other location, without the requirements imposed on a Technical Support Center (TSC). |
| (ii) ~~For nuclear power reactor licensees, a licensee onsite operational support center;~~ | NUREG-0696, “Functional Criteria for Emergency Response Facilities,” provides that the operational support center (OSC) is an onsite area separate from the control room and the TSC where licensee operations support personnel will assemble in an emergency. For a defueled power plant, an OSC is no longer required to meet its original purpose of an assembly area for plant logistical support during an emergency. The OSC function can be incorporated into another facility. |
| b. ~~For a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, either a facility located between 10 miles and 25 miles of the nuclear power reactor site(s), or a primary facility located less than 10 miles from the~~ | See basis for 50.47(b)(3). |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| ~~nuclear power reactor site(s) and a backup facility located between 10 miles and 25 miles of the nuclear power reactor site(s). An emergency operations facility may serve more than one nuclear power reactor site. A licensee desiring to locate an emergency operations facility more than 25 miles from a nuclear power reactor site shall request prior Commission approval by submitting an application for an amendment to its license. For an emergency operations facility located more than 25 miles from a nuclear power reactor site, provisions must be made for locating NRC and offsite responders closer to the nuclear power reactor site so that NRC and offsite responders can interact face-to-face with emergency response personnel entering and leaving the nuclear power reactor site. Provisions for locating NRC and offsite responders closer to a nuclear power reactor site that is more than 25 miles from the emergency operations facility must include the following:~~<br><br>(1)~~ Space for members of an NRC site team and Federal, State, and local responders;~~<br><br>(2)~~ Additional space for conducting briefings with emergency response personnel;~~<br><br>(3)~~ Communication with other licensee and offsite emergency response facilities;~~<br><br>(4)~~ Access to plant data and radiological information; and~~<br><br>(5)~~ Access to copying equipment and office supplies;~~ | |
| c.~~ By June 20, 2012, for a nuclear power reactor licensee's emergency operations facility required by paragraph 8.a of this section, a facility having the following capabilities:~~<br><br>(1)~~ The capability for obtaining and displaying plant data and radiological information for each reactor at a nuclear power reactor site and for each nuclear power reactor site that~~ | See basis for 50.47(b)(3). |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| ~~the facility serves;~~<br><br>(2)~~ The capability to analyze plant technical information and provide technical briefings on event conditions and prognosis to licensee and offsite response organizations for each reactor at a nuclear power reactor site and for each nuclear power reactor site that the facility serves; and~~<br><br>(3)~~ The capability to support response to events occurring simultaneously at more than one nuclear power reactor site if the emergency operations facility serves more than one site; and~~ | |
| d.~~ For nuclear power reactor licensees, an alternative facility (or facilities) that would be accessible even if the site is under threat of or experiencing hostile action, to function as a staging area for augmentation of emergency response staff and collectively having the following characteristics: the capability for communication with the emergency operations facility, control room, and plant security; the capability to perform offsite notifications; and the capability for engineering assessment activities, including damage control team planning and preparation, for use when onsite emergency facilities cannot be safely accessed during hostile action. The requirements in this paragraph 8.d must be implemented no later than December 23, 2014, with the exception of the capability for staging emergency response organization personnel at the alternative facility (or facilities) and the capability for communications with the emergency operations facility, control room, and plant security, which must be implemented no later than June 20, 2012.~~ | See basis for section IV.1. regarding hostile action. |
| e.~~ A licensee shall not be subject to the requirements of paragraph 8.b of this section for an existing emergency operations facility approved as of December 23, 2011;~~ | See basis for 50.47(b)(3). |
| 9.a. Provisions for communications with contiguous State/local governments ~~within the plume exposure pathway EPZ~~. Such communication shall be tested monthly. | See basis for 50.47(b) and (b)(10).<br><br>The State and the local governments in which the nuclear facility is located need to be informed of events and emergencies, so lines of communication must be maintained. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| 9.c. Provision for communications among the ~~nuclear power reactor control room, the onsite technical support center, and the emergency operations facility; and among the~~ nuclear facility, the principal State and local emergency operations centers, ~~and the field assessment teams~~. Such communications systems shall be tested annually. | Because of the low probability of design-basis accidents or other credible events that would be expected exceed the EPA PAGs and the available time for event mitigation, there is no need for the TSC, EOF or field assessment teams.<br><br>Also see justification for 50.47(b)(3).<br><br>Communication with State and local EOCs is maintained to coordinate assistance on site if required. |
| 9.d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the ~~nuclear power reactor control room, the onsite technical support center, and the emergency operations~~ facility. Such communications shall be tested monthly. | The functions of the control room, EOF, TSC and OSC may be combined into one or more locations due to the smaller facility staff and the greatly reduced required interaction with State and local emergency response facilities.<br><br>Also see basis for 50.47(b). |
| **10 CFR Part 50, Appendix E, section IV.F** | **Basis for Change** |
| 1. The program to provide for: (a) The training of employees and exercising, by periodic drills, of radiation emergency plans to ensure that employees of the licensee are familiar with their specific emergency response duties, and (b) The participation in the training and drills by other persons whose assistance may be needed in the event of a radiation emergency shall be described. This shall include a description of specialized initial training and periodic retraining programs to be provided to each of the following categories of emergency personnel:<br><br>i. Directors and/or coordinators of the plant emergency organization;<br><br>ii. Personnel responsible for accident assessment, including control room shift personnel;<br><br>iii. Radiological monitoring teams;<br><br>iv. Fire control teams (fire brigades);<br><br>v. Repair and damage control teams;<br><br>vi. First aid and rescue teams;<br><br>vii. Medical support personnel;<br><br>viii. ~~Licensee's headquarters support~~ | The number of staff at decommissioning sites is generally small but is commensurate with the need to safely store spent fuel at the facility in a manner that is protective of public health and safety. Decommissioning sites typically have a level of emergency response that does not require additional response by headquarters personnel. Therefore, the staff considers exempting licensee's headquarters personnel from training requirements reasonable. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| ~~personnel;~~<br><br>ix. Security personnel.<br><br>In addition, a radiological orientation training program shall be made available to local services personnel; e.g., local emergency services~~/Civil Defense~~, local law enforcement personnel~~, local news media persons~~. | |
| 2. The plan shall describe provisions for the conduct of emergency preparedness exercises as follows: Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, ~~test the public alert and notification system,~~ and ensure that emergency organization personnel are familiar with their duties. | Because of the low probability of design-basis accidents or other credible events that would be expected to exceed the limits of EPA PAGs and the available time for event mitigation, the public alert and notification system will not be used and therefore requires no testing.<br>Also see basis for 50.47(b) |
| a.~~ A full participation exercise which tests as much of the licensee, State, and local emergency plans as is reasonably achievable without mandatory public participation shall be conducted for each site at which a power reactor is located. Nuclear power reactor licensees shall submit exercise scenarios under § 50.4 at least 60 days before use in a full participation exercise required by this paragraph 2.a.~~<br><br>F.2.a.(i), (ii), and (iii) are not applicable. | Since the need for off-site emergency planning is relaxed due to the low probability of design-basis accidents or other credible events that would be expected to exceed the limits of EPA PAGs and the available time for event mitigation, no off-site emergency plans are in place to test.<br><br>The intent of submitting exercise scenarios at power reactors is to check that licensees utilize different scenarios in order to prevent the preconditioning of responders at power reactors. For defueled sites, there are limited events that could occur and the previously routine progression to General Emergency in power reactor site scenarios is not applicable to a decommissioning site.<br>The licensee is exempt from F.2.a.(i)-(iii) because the licensee is exempt from the umbrella provision of F.2.a. |
| b. Each licensee at each site shall conduct a subsequent exercise of its onsite emergency plan every 2 years. ~~Nuclear power reactor licensees shall submit exercise scenarios under § 50.4 at least 60 days before use in an exercise required by this paragraph 2.b. The exercise may be included in the full participation biennial exercise required by paragraph 2.c. of this section.~~ In addition, the licensee shall take actions necessary to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional | See basis for section IV.F.2.a.<br><br>The low probability of design-basis accidents or other credible events that would exceed the EPA PAGs and the available time for event mitigation at a decommissioning site render TSCs, OSCs and EOFs unnecessary. The principal functions required by regulation can be performed at an onsite location that does not meet the requirements of the TSC, OSC or EOF. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| areas of the licensee's onsite emergency response capabilities. The principal functional areas of emergency response include activities such as management and coordination of emergency response, accident assessment, event classification, notification of offsite authorities, and assessment of the onsite ~~and offsite~~ impact of radiological releases~~, protective action recommendation development, protective action decision making, plant~~ system repair and mitigative action implementation. During these drills, activation of all of the licensee's emergency response facilities ~~(Technical Support Center (TSC), Operations Support Center (OSC), and the Emergency Operations Facility (EOF))~~ would not be necessary, licensees would have the opportunity to consider accident management strategies, supervised instruction would be permitted, operating staff in all participating facilities would have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills may focus on the onsite exercise training objectives. | |
| c.~~ Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the radiological response plan. Where the offsite authority has a role under a radiological response plan for more than one site, it shall fully participate in one exercise every two years and shall, at least, partially participate in other offsite plan exercises in this period. If two different licensees each have licensed facilities located either on the same site or on adjacent, contiguous sites, and share most of the elements defining co-located licensees, then each licensee shall:~~<br><br>(1)~~ Conduct an exercise biennially of its onsite emergency plan;~~<br><br>(2)~~ Participate quadrennially in an offsite biennial full or partial participation exercise;~~<br><br>(3)~~ Conduct emergency preparedness activities and interactions in the years between its participation in the offsite full or partial participation exercise with offsite authorities, to~~ | See basis for section IV.F.2a. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| ~~test and maintain interface among the affected State and local authorities and the licensee. Co-located licensees shall also participate in emergency preparedness activities and interaction with offsite authorities for the period between exercises;~~<br><br>(4)~~ Conduct a hostile action exercise of its onsite emergency plan in each exercise cycle; and~~<br><br>(5)~~ Participate in an offsite biennial full or partial participation hostile action exercise in alternating exercise cycles.~~ | |
| d.~~ Each State with responsibility for nuclear power reactor emergency preparedness should fully participate in the ingestion pathway portion of exercises at least once every exercise cycle. In States with more than one nuclear power reactor plume exposure pathway EPZ, the State should rotate this participation from site to site. Each State with responsibility for nuclear power reactor emergency preparedness should fully participate in a hostile action exercise at least once every cycle and should fully participate in one hostile action exercise by December 31, 2015. States with more than one nuclear power reactor plume exposure pathway EPZ should rotate this participation from site to site.~~ | See basis for section IV.2. |
| e. Licensees shall enable any State or local Government ~~located within the plume exposure pathway EPZ~~ to participate in the licensee's drills when requested by such State or local Government. | See basis for section IV.2. |
| f. Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC~~, in consultation with FEMA,~~ cannot (1) find reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency or (2) determine that the Emergency Response Organization (ERO) has maintained key skills specific to emergency response. ~~The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercises.~~ | The U.S. Federal Emergency Management Agency (FEMA) is responsible for the evaluation of an offsite response exercise. No action is expected from State or local government organizations in response to an event at a decommissioning site other than firefighting, law enforcement and ambulance/medical services. Memoranda of understanding should be in place for those services. Offsite response organizations will continue to take ad hoc actions to protect the health and safety of the public as they would at any other industrial site. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| i. Licensees shall use drill and exercise scenarios that provide reasonable assurance that anticipatory responses will not result from preconditioning of participants. ~~Such scenarios for nuclear power reactor licensees must include a wide spectrum of radiological releases and events, including hostile action.~~ Exercise and drill scenarios as appropriate must emphasize coordination among onsite and offsite response organizations. | For defueled sites, there are limited events that could occur and the previously routine progression to General Emergency in power reactor site scenarios is not applicable to a decommissioning site. Therefore the licensee is not expected to demonstrate response to a wide spectrum of events.<br><br>Also see basis for section IV.1 regarding hostile action. |
| j.~~ The exercises conducted under paragraph 2 of this section by nuclear power reactor licensees must provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to implement the principal functional areas of emergency response identified in paragraph 2.b of this section. Each exercise must provide the opportunity for the ERO to demonstrate key skills specific to emergency response duties in the control room, TSC, OSC, EOF, and joint information center. Additionally, in each eight calendar year exercise cycle, nuclear power reactor licensees shall vary the content of scenarios during exercises conducted under paragraph 2 of this section to provide the opportunity for the ERO to demonstrate proficiency in the key skills necessary to respond to the following scenario elements: hostile action directed at the plant site, no radiological release or an unplanned minimal radiological release that does not require public protective actions, an initial classification of or rapid escalation to a Site Area Emergency or General Emergency, implementation of strategies, procedures, and guidance developed under § 50.54(hh)(2), and integration of offsite resources with onsite~~ justification~~. The licensee shall maintain a record of exercises conducted during each eight year exercise cycle that documents the content of scenarios used to comply with the requirements of this paragraph. Each licensee shall conduct a hostile action exercise for each of its sites no later than December 31, 2015. The first eight-year exercise cycle for a site will begin in the calendar year in which the first hostile action exercise is conducted. For a site licensed under Part 52, the first eight-year exercise cycle begins in the calendar year of the initial exercise required by Section~~ | See basis for section IV.F.2. |

**Table 1**
**EXEMPTIONS FOR CONSIDERATION**

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| ~~IV.F.2.a.~~ | |
| **10 CFR Part 50, Appendix E, section IV.I** | **Basis for Change** |
| ~~By June 20, 2012, for nuclear power reactor licensees, a range of protective actions to protect onsite personnel during hostile action must be developed to ensure the continued ability of the licensee to safely shut down the reactor and perform the functions of the licensee's emergency plan.~~ | See basis for section IV.E.d. |

## References


1) 10 CFR 50.47, “Emergency Plans.”

2) 10 CFR 50.54, “Conditions of Licenses.”

3) 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities.”

4) 10 CFR Part 72.32, “Emergency Plan”

5) U.S. Nuclear Regulatory Commission, “Integrated Rulemaking Plan for Nuclear Power Plant Decommissioning,” Commission Paper SECY-00-0145, June 28, 2000 Agencywide Document Access and Management System (ADAMS) Accession No. ML003721626.

6) U.S. Nuclear Regulatory Commission, “Policy Issues Related to Safeguards, Insurance, and Emergency Preparedness Regulations at Decommissioning Nuclear Power Plants Storing Fuel in Spent Fuel Pools” Commission Paper SECY-01-0100, ADAMS Accession No. ML011450420.

7) U.S. Nuclear Regulatory Commission, “Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants” NUREG-1738 February 2001 ADAMS Accession No. ML010430066.

8) Memorandum from William Travers to Commission re: Status of Regulatory Exemptions for Decommissioning Plants”, August, 2002 ADAMS Accession No. ML030550706.

9) U.S. Nuclear Regulatory Commission, “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor” (Draft Report for Comment) June 2013, ADAMS Accession No. ML13133A132.

10) U.S. Nuclear Regulatory Commission, “A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants” NUREG/CR-6451 August 1997, ADAMS Accession No. ML082260098.

11) U.S. Nuclear Regulatory Commission, NUREG-1864, “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant” ADAMS Accession No. ML071340012.

12) U.S. Nuclear Regulatory Commission, Commission Paper SECY-97-120,“Rulemaking Plan for Emergency Planning Requirements for Permanently Shutdown Nuclear Power Plant Sites”, July 1997, ADAMS Accession No. ML003752513.

13) NEI 99-01, Revision 6 (Draft) “Methodology for the Development of Emergency Action Levels for Non-Passive Reactors”, November 2012, ADAMS Accession No. ML12326A805.

**Attachment 1**
**GUIDANCE FOR EVALUATION OF DECOMMISSIONING EMERGENCY PLANS**

The following guidance should be used for the review of Defueled Emergency Plans for sites undergoing decommissioning:

## 1.0 Emergency Response Equipment and Facilities

Applicable Requlation(s): 10 CFR 50.47(b)(8) and (9), Appendix E to 10 CFR Part 50, Section IV.E

### 1.1. Back ground and Discussion

Operating power reactor sites require separate facilities for functions of evaluation and coordination of activities associated with the emergency, technical support, plant operation, assembly of logistical support personnel, and dissemination of information. When a site enters decommissioning, most of the plant systems are no longer required for operation or for mitigation of an accident. Most of the design basis accidents are no longer credible. The staff required to support the site is also much smaller. Facility functions may also be combined, and therefore, physical locations may be eliminated.

### 1.2. Guidance

The emergency plan should describe the onsite equipment and facilities designated for use during emergencies. The plan should describe the principal and alternate locations from which emergency control and assessment activities will occur. At least one location should be habitable during any emergency.

The emergency plan should include the means for identifying a command center to be used in an emergency. The criteria for evacuating a command center and re-establishing control from an alternate location should also be described. The plan should identify one or more locations from which licensee emergency workers would be dispatched to perform radiation surveys, damage assessment, emergency repair, or other mitigating tasks.

The protective equipment and supplies available to emergency response personnel should be described. Types of equipment and supplies may include:

- individual respiratory equipment, including self-contained breathing apparatus
- protective clothing
- firefighting equipment and gear
- supplemental lighting
- medical supplies
- contamination control and decontamination equipment
- communications equipment
- radiation detection equipment, including radiation meters, air samplers, dosimeters
- hazardous material detection equipment
- potassium iodide

The emergency plan should include criteria for issuing respiratory equipment, locations of emergency equipment and supplies, means for distributing these items and criteria for dispensing potassium iodide, if required.

The emergency plan should also include inventory lists indicating the emergency equipment and supplies provided at specified locations. The plan should describe the primary and alternate onsite and offsite communication systems that would be used to transmit and receive information throughout the emergency. A backup means of offsite communication to a commercial telephone should be provided for notification of emergencies and requests for assistance.

## 2.0 Staffing and Communication

Applicable Requlation(s): 10 CFR 50.47(b)(1), (2), (5) and (6), Appendix E to 10 CFR Part 50, Sections IV.A, C and D

### 2.1. Background and Discussion:

Table B-1 in NUREG-0654/FEMA-REP-1, Revision 1 describes the minimum emergency response staffing requirements for nuclear power plant licensed per 10 CFR Part 50 and 10 CFR Part 52. The staff recognizes that due to the limited number, lower possible frequency and relative magnitude of events at a defueled facility, fewer staff may be required during decommissioning. The major functional areas remain the same, but the major tasks are different and the time available to take mitigating actions changes significantly. Defueled Technical Specifications typically will define the onshift operating staff at a defueled decommissioning site as two positions: a certified fuel handler and an operator or technician. The major responsibility of the onshift staff, while there is fuel in the SFP, is to maintain SFP cooling. Performing the role of an Emergency Director should be within the qualifications and capabilities of the designated onshift staff member.

### 2.2. Guidance

#### 2.2.1 Responsibilities

The emergency plan should describe the emergency organization to be activated onsite for possible events, and offsite augmentation and support. The plan should delineate the authorities and responsibilities of key positions and groups, and identify the communication chain for notifying and mobilizing personnel during normal and non-working hours. Personnel with the responsibility for event classification, onsite protective action decisions, and prompt notification of State and local government authorities and the NRC should be identified.

#### 2.2.2 Decommissioning Facility Organization

The emergency plan should provide a brief description of the normal (day-to-day) facility organization and identify by position those with responsibility to declare an emergency and to initiate the appropriate response. Personnel responsible for maintaining the emergency plan and emergency response procedures should be identified.

2.2.3 Onsite Emergency Response Organization

The emergency plan should identify the onsite emergency response organization for the facility, including during periods such as holidays, weekends, and extended periods when normal operations are not being conducted. Organizational charts and tables should be used when appropriate. If the organization is activated in phases, the plan should describe the base organization and each additional component that may be activated to augment the organization. Typically, a minimum staff to augment the minimum onshift staff is manned within an hour of declaration of an Alert with a goal of total augmentation within two hours. The plan should clearly state the minimum level of staffing needed to effectively implement the plan for each period or phase described.

2.2.4 Direction and Coordination

The emergency plan should designate the position of the person, and alternate(s), who has principal responsibility for implementing and directing the emergency response. This person's duties and authorities would include:

- control of the situation
- initial classification, escalation or termination of the emergency condition
- event notification
- coordination of the staff and offsite personnel who augment the staff
- communication with parties requesting information regarding the event
- onsite protective measure decision-making
- request of support from offsite agencies

The emergency plan should also describe this person's authority to delegate responsibilities and the individuals who may be delegated certain emergency responsibilities.

2.2.5 Onsite Staff Emergency Assignments

The emergency plan should specify the organizational group or groups assigned to the functional areas of emergency activity listed below. The plan should also describe strategies for staffing these positions if the emergency lasts for an extended period of time. The duties, authorities, and interface with other groups and offsite assistance should be described. The organizational groups should provide support in the following areas:

- facility systems operations,
- fire control,
- onsite protective measures, including personnel evacuation and accountability,
- search and rescue operations,
- first aid,
- communications,
- onsite radiological survey and assessment,
- personnel and facility decontamination,
- facility security and access control,
- facility repair and damage control,
- post-event assessment,

- record keeping,
- media contact, and
- criticality safety assessment

2.2.6 Emergency Response Records

The emergency plan should describe the assignment of responsibility for reporting and recording incidents of abnormal operation, equipment failure, and accidents that led to a facility emergency. Decommissioning records shall be maintained until the license is terminated as required by10 CFR 50.75(g). Records of an emergency or incident to be maintained should include the following:

- cause of the incident,
- personnel and equipment involved,
- extent of injury and damage (onsite and offsite) as a result of the incident,
- locations of contamination with the final decontamination survey results,
- corrective actions taken to terminate the emergency,
- actions taken or planned to prevent a recurrence of the incident,
- onsite and offsite assistance requested and received, and
- any program changes resulting from a critique of emergency response activities.

The emergency plan should provide a description of the records associated with emergency plan maintenance that will be kept. These should include the following:

- training and retraining (including lesson plans and test questions),
- drills, exercises, and related critiques,
- inventory and locations of emergency equipment and supplies,
- maintenance, surveillance, calibration, and testing of emergency equipment and supplies,
- letters of agreement with offsite support organizations,
- reviews and updates of the emergency plan submitted per 10 CFR Part 50.54(q), and
- notification of onsite personnel and offsite response organizations affected by an update of the plan or its implementing procedures

The emergency plan should include provisions for an annual review and audit of the emergency preparedness program to ensure the program remains adequate. Elements of the audit should include a review of the following:

- emergency plan and associated procedures,
- emergency response training activities,
- records of emergency facilities, equipment, and supplies,
- records associated with offsite response agencies interface (such as training and letters of agreement),
- exercises, drills, communications, and inventory checks, and
- activation of the emergency plan since the last audit

2.2.7 Coordination with Offsite Response Organizations

The emergency plan should identify the principal State agency and other government (local, county, State, and Federal) agencies or organizations having authority for radiological or other hazardous material emergencies. The agencies' and/or organization's location and specific response capabilities in terms of personnel and resources should be described. The plan should include a description of the onsite and offsite services that support emergency response operations, including the following:

- decontamination facilities,
- medical treatment facilities,
- first aid personnel,
- fire fighters,
- law enforcement assistance, and
- ambulance services

2.2.8 Notification and Coordination

The emergency plan should describe the means used to activate the emergency response organization for each class of emergency on a 24-hours per day/7-days per week basis. The plan should describe the means provided to detect and notify the licensee's onshift staff of any abnormal conditions or of any danger to safe operations (e.g., a severe weather warning). The means to promptly notify State and local government authorities and the NRC should be described. The ability to request offsite assistance, including medical assistance for the treatment of contaminated injured onsite workers, should also be described. The plan should include the commitment to notify the NRC Operations Center immediately after notification of State and local government authorities but no later than one hour after an emergency is declared.

2.2.9 Information to be Communicated

The emergency plan should describe the type of information to be communicated to State and local government authorities and the NRC. The information should be clear, concise and should avoid technical terms and jargon. The types of information to be communicated should include the status of the facility, if a release of radioactive material is occurring or could occur, and dose rate projections. A standard reporting checklist should be included in the plan to facilitate timely notification for each postulated accident.

## 3.0 <u>Mitigation of Consequences</u>

Applicable Regulation(s): 10 CFR 50.47(b)(3), (8) and (10), Appendix E to 10 CFR Part 50, Section IV.B

### 3.1. Background and Discussion

Sites which hold spent fuel susceptible to zirconium fires have been exempted from some EP regulations based on their analysis showing the ability to perform actions to prevent such events or to take offsite protective actions were necessary. A site-specific SFP analysis should show that there is sufficient time from the loss of SFP inventory

until the onset of a zirconium fire to take the actions to mitigate the inventory loss and prevent a zirconium fire and to take offsite protective actions. Specifically, a time of at least ten hours from the loss of SFP inventory, without air cooling, to a temperature of 900 degrees C should be one conclusion from this site specific analysis. The emergency plan should describe the equipment, personnel, resources, such as water supplies, procedures and strategies in place for movement of any necessary portable equipment, initial and continuing training, that will be relied upon for prevention of a zirconium fire in the SFP. These mitigative strategies may have been developed as part of a response to or the result of NRC Order on Mitagative Strategies (EA-12-049). A time estimate for completing necessary actions to preclude the zirconium fire should be made.

## 3.2. Guidance

### 3.2.1 Limiting Actions

The emergency plan should describe the means and equipment provided for limiting the consequences of each type of accident identified in the plan. The plan should address the actions and systems in place to reduce the magnitude and/or reduce the effect of a radioactive or hazardous material release that has occurred. The plan should include actions to be taken to limit and mitigate the consequences to the public and workers. Means for limiting releases could include the following:

- sprinkler systems and other fire suppression systems
- fire detection systems
- firefighting capabilities
- filtration or holdup systems
- use of water sprays on airborne releases of radioactive material
- automatic shut-off of process or ventilation flow
- use of fire-resistant building materials

If portable equipment is used to prevent or mitigate events, the emergency plan should describe the procedures, storage and maintainability of that equipment.

Based upon the type of emergency, the emergency plan should describe the criteria for the shutdown of systems or the facility and any steps to be taken to ensure a safe, orderly shutdown of fuel handling operations and the approximate time required to complete the shutdown.

### 3.2.2 Onsite Protective Actions

The emergency plan should describe the nature of onsite protective actions, criteria for implementing those actions, the areas involved, and the procedures for notification to potentially affected persons. The plan should allow for timely relocation of onsite persons, effective use of protective equipment and supplies, and use of appropriate contamination control measures. The plan should describe the means for controlling and/or minimizing radiological exposures for personnel onsite, and any personnel expected to arrive onsite. The onsite exposure guidelines should be consistent with the EPA PAGs to be used in actions to control fires, stop releases, or protect the facilities. Exposure guidelines should be provided for:

- search and rescue
- removing injured persons
- undertaking mitigating actions
- performing assessment actions
- providing onsite first aid
- performing personnel decontamination
- providing ambulance service or offsite medical treatment

The emergency plan should include methods for onsite personnel evacuation and accountability. This could include:

- criteria for ordering a site evacuation
- means and timely notification of onsite persons impacted
- provisions for determining and maintaining accountability of assembled and evacuated personnel, and for identifying and determining the locations of personnel that were not evacuated
- search and rescue
- locations of onsite and offsite assembly areas
- evacuation routes and means for transporting onsite personnel (e.g., privately owned vehicles, buses, company vehicles)
- monitoring of evacuees for contamination and control measures if contamination is found
- criteria for command center and assembly area evacuation and re-establishment at an alternate location
- means for evacuating and treating onsite injured personnel, including potentially contaminated personnel

The emergency plan should describe provisions for preventing further spread of radioactive materials and for minimizing personnel exposure from radioactive materials. The plan should specify action levels for decontaminating personnel. The plan should describe provisions for determining the doses and dose commitments from external radiation exposure and internally deposited radioactive material received by emergency response personnel, including personnel from offsite emergency response organizations (e.g. fire, medical, police).

The emergency plan should describe arrangements made for hospital and medical services, both primary and backup, and their capabilities to evaluate and treat contaminated, injured persons, and injuries involving radiation, radioactive materials, and other hazardous materials used in conjunction with radioactive materials. The medical facility description should include capabilities to control any contamination that may be associated with the physical injuries. The plan should specify how injured personnel who are potentially contaminated will be transported to offsite medical facilities. The plan should describe how chemicals or hazardous materials stored onsite may impact transporting injured personnel. The commitment to provide ambulance and hospital personnel with health physics support should be included.

3.2.3 Assessment of Releases

The emergency plan should discuss the actions to be taken to determine the extent of the problem and to decide what corrective actions may be required for each class of emergency. This should include the types and methods of onsite and offsite sampling and monitoring in case of a release of radioactive or other hazardous material. The provisions for projection of offsite radiation exposures should be described.

## 4.0 Emergency Action Levels

Applicable Regulation(s): 10 CFR 50.47(b)(4), Appendix E to 10 CFR Part 50, Section IV.B, 10 CFR 72.32.a.

### 4.1. Background and Discussion

Recognition Category Permanently Defueled (PD) of the Nuclear Energy Institute (NEI) document NEI 99-01 Revision 6, “Methodology for Development of Emergency Action Levels," provides a stand-alone set of initiating conditions (ICs) and emergency action levels (EALs) for a permanently defueled NPP to consider for use in developing a site-specific emergency classification scheme. For development, it was assumed that the plant had operated under a 10 CFR Part 50 license and that the operating company has permanently ceased plant operations. Further, the licensee intends to store the spent fuel within the plant for some period of time. When in a permanently defueled condition, the licensee will typically receive approval from the NRC for exemption from specific emergency planning requirements. These exemptions reflect the lower radiological source term and risks associated with spent fuel pool storage relative to an operating power reactor. Source terms and accident analyses associated with plausible accidents are documented in the station’s Final Safety Analysis Report (FSAR), as updated. As a result, each licensee will need to develop a site-specific emergency classification scheme using the NRC-approved exemptions, revised source terms, and revised accident analyses as documented in the station’s FSAR.

Recognition Category PD uses the same emergency classification levels (ECLs) as operating reactors; however, the source term and accident analyses typically limit the ECLs to an Unusual Event and Alert. The Unusual Event ICs provide for an increased awareness of abnormal conditions while the Alert ICs are specific to actual or potential impacts to spent fuel. The source terms and release motive forces associated with a permanently defueled plant would not be sufficient to require declaration of a Site Area Emergency or General Emergency unless a zirconium fire occurs.

A permanently defueled station is essentially a spent fuel storage facility with the spent fuel stored in a pool of water that serves as both a cooling medium (i.e., removal of decay heat) and a shield from direct radiation. These primary functions of the spent fuel storage pool are the focus of the Recognition Category PD ICs and EALs. Radiological effluent IC and EALs were included to provide a basis for classifying events that cannot be readily classified based on an observable event or plant conditions alone.

Appropriate ICs and EALs from the other Recognition Categories of NEI 99-01 were modified and included in Recognition Category PD to address a spectrum of the events that may affect a spent fuel pool. The Recognition Category PD ICs and EALs reflect

the relevant guidance in this document (e.g., the importance of avoiding both over-classification and under-classification). Nonetheless, each licensee will need to develop its emergency classification scheme using the NRC-approved exemptions, and the source terms and accident analyses specific to the licensee. Security-related events will also need to be considered and documented in the licensee Physical Security Plan and written implementing procedures.

Selected guidance in NEI 99-01 is applicable to licensees electing to use their 10 CFR Part 50 emergency plan to fulfill the requirements of 10 CFR 72.32 for a stand-alone Independent Spent Fuel Storage Installation (ISFSI). The emergency classification levels applicable to an ISFSI are consistent with the requirements of 10 CFR Part 50 and the guidance in NUREG 0654/FEMA-REP-1. The initiating conditions germane to a 10 CFR 72.32 emergency plan (as described in NUREG-1567) are subsumed within the classification scheme for a 10 CFR 50.47 emergency plan.

The generic ICs and EALs for an ISFSI are presented in NEI 99-01, ISFSI ICs/EALs. IC E-HU1 covers the spectrum of credible natural and man-made events included within the scope of an ISFSI design. This IC is not applicable to installations or facilities that may process and/or repackage spent fuel (e.g., a Monitored Retrievable Storage Facility (MRS) or an ISFSI at a spent fuel processing facility). In addition, appropriate aspects of IC HU1 and IC HA1 should also be included to address security events directed against an ISFSI.

## 4.2. Guidance

### 4.2.1 Unusual Event

The emergency plan should identify events which could lead to initiation of an Unusual Event. Initiating events may include:

- release of gaseous or liquid radioactivity greater than 2 times the (site-specific effluent release controlling document) limits for 60 minutes or longer
- unplanned rise in plant radiation levels
- unplanned spent fuel pool temperature rise
- confirmed security condition or threat
- hazardous event affecting safety system equipment necessary for spent fuel cooling
- other conditions exist which in the judgment of the Emergency Director warrant declaration of an Unusual Event

### 4.2.2 Alert

The emergency plan should identify events which could lead to initiation of an Alert. Initiating events may include:

- release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem total effective dose equivalent (TEDE) or 50 mrem thyroid committed dose equivalent (CDE)
- unplanned rise in plant radiation levels that impedes plant access required to maintain spent fuel integrity

- hostile action within the Owner Controlled Area or airborne attack threat within 30 minutes
- other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert

4.2.3 Independent Spent Fuel Storage Installation

If the licensee elects to transfer the spent fuel and store it in an ISFSI, the emergency plan should also identify events for the ISFSI which could lead to initiation of an Unusual Event. Initiating events may include:

- Damage to a loaded cask confinement boundary

## 5.0 Exercises

The emergency plan should describe the provisions for periodic drills and exercises. Communications checks with offsite agencies,and radiological/health physics, medical, and fire drills should be performed at the interval established by 10 CFR 72.32(a) or (b). The biennial onsite exercise should test the effectiveness of the personnel, plan and procedures, and readiness of facilities, equipment, supplies and instrumentation. Offsite responses organizations should be invited to participate, however, participation is not required. The plan should describe the responsibility for developing the exercise accident scenario, requirements for non-participating observers to evaluate the effectiveness of the exercise, the need for a critique of the exercise, and if deficiencies are found, how they will be corrected.

## 6.0 Assistance

The emergency plan should describe provisions and arrangements for assistance from offsite response organizations during and after an emergency. The plan should indicate the location of local assistance with respect to the facility. Exposure guidelines should be clearly communicated to offsite emergency response personnel. The plan should identify the services to be performed, means of communication and notification, and types of agreements that are in place for the following:

- medical treatment facilities,
- first aid personnel and/or ambulance service,
- fire fighters, and
- local law enforcement assistance/documented memorandum of agreements (specific details may be Safeguards Information.

The emergency plan should describe the measures that will be taken to ensure that offsite response organizations maintain an awareness of their respective roles in an emergency and have the necessary equipment, supplies and periodic training to carry out their emergency response functions. Any provisions to suspend security or safeguards measures for site access during an emergency should be described.

The licensee should offer to meet at least annually with each offsite response organization providing onsite support as identified in the licensee's emergency plan, to review items of mutual interest, including relevant changes to the emergency plan. The

licensee should discuss the emergency action level scheme, notification procedures, and overall response coordination process during these meetings.

**Attachment 2**
**PREVIOUSLY APPROVED LICENSING ACTIONS**

| Licensee | Date Operations Ceased | Date Exemption Issued | Basis for Exemption |
|---|---|---|---|
| Humbolt Bay | 7/2/76 | 4/29/87 | The staff evaluated offsite radiological consequences of potential accidents involving the fuel stored in the spent fuel pool including a fuel handling accident, a non-mechanistic heavy load drop, and a seismically- or otherwise-induced rearrangement of the stored fuel assemblies. Other hypothetical accident scenarios considered by the staff were a non-mechanistic expulsion of all pool water to the atmosphere, a spent fuel rupture, and uncontrolled release of all contents of the liquid radwaste tanks to the discharge canal. The staff concluded that all atmospheric releases were well below EPA PAGs. |
| La Crosse | 4/30/87 | 7/8/88 | The staff evaluated the offsite consequences of potential accidents to the fuel stored in the spent fuel pool. The analysis assumed all fuel rods damaged with no iodine filters operating, and no fuel pool water missing. In this scenario, the doses at the exclusion area boundary would be less than 25% of the 10 CFR Part 100 paragraph 11 guideline values, i.e., much less than 75 rem for the thyroid and 6 rem for whole-body dose. The above dose values are the acceptance criteria value from the NRC Standard Review Plan (NUREG-800) Section 15.7.5 on spent fuel cask drop accidents. Similarly, the calculated doses are well below EPA PAGs. |
| Fort St. Vrain | 8/18/89 | 12/31/90 | Analyzed radiological consequences of potential accidents involving a fuel handling accident ( i.e., dropped fuel shipping cask) provided doses offsite less than EPA PAGs. |
| Rancho Seco | 6/7/89 | 2/22/91 | Analyzed radiological consequences of potential accidents involving a fuel handling accident ( i.e., dropped fuel shipping cask) provide doses offsite less than EPA PAGs. |

**Attachment 2**
**PREVIOUSLY APPROVED LICENSING ACTIONS**

| Licensee | Date Operations Ceased | Date Exemption Issued | Basis for Exemption |
|---|---|---|---|
| Yankee Rowe | 10/1/91 | 10/30/92 | Analyzed radiological consequences of potential accidents involving a fuel handling accident ( i.e., dropped fuel shipping cask) provide doses offsite less than EPA PAGs. |
| Trojan | 11/2/92 | 9/30/93 | Analyzed radiological consequences of potential accidents involving a fuel handling accident ( i.e., dropped fuel shipping cask) provide doses offsite less than EPA PAGs.<br><br>The staff concluded that in view of the low likelihood of a seismic event > 0.5g and the time elapsed since shutdown of the facility, and the configuration of the fuel in the spent fuel pool, that there would be sufficient time after a postulated loss of water and before the initiation of a cladding fire for the licensee to implement actions to preclude heat up of the spent fuel. |

**Attachment 2**
**PREVIOUSLY APPROVED LICENSING ACTIONS**

| Licensee | Date Operations Ceased | Date Exemption Issued | Basis for Exemption |
|---|---|---|---|
| Haddam Neck | 7/22/96 | 8/28/98 | The staff evaluated:<br><br>1. Release of activity from combustible ion exchanger resin and fuel handling accidents would not exceed EPA PAGs.<br>2. For gamma radiation due to a loss of spent fuel pool level, it would take 2.6 days to exceed EPA PAGs.<br>3. For a bounding scenario where the fuel is totally uncovered, the decay heat would not heat up higher than 565 degrees Celsius (C); therefore the cladding would stay intact.<br><br>The staff concluded that the postulated doses to the general public from any reasonably conceivable accident would not exceed EPA PAGs and, for the loss of fuel pool level, the length of time available gives confidence that mitigative actions could be taken and provides confidence that additional offsite measures could be taken without planning. |
| Maine Yankee | 12/6/96 | 9/3/98 | The staff evaluated:<br><br>1. A fire involving resin and gamma radiation due to a loss of spent fuel pool level not exceeding EPA PAGs.<br>2. A bounding scenario where the fuel is totally uncovered and no natural circulation flow path exists. The staff calculated that it would take ~10 hours to heat up to 900 degrees C.<br><br>The staff concluded that the postulated doses to the general public from any reasonably conceivable accident would not exceed EPA PAGs and, for the bounding accident, the length of time available gives confidence that mitigative actions and, if necessary, offsite measures for the public could be taken without preplanning. |

**Attachment 2**
**PREVIOUSLY APPROVED LICENSING ACTIONS**

| Licensee | Date Operations Ceased | Date Exemption Issued | Basis for Exemption |
|---|---|---|---|
| Big Rock Point | 8/29/97 | 9/30/98 | The staff evaluated:<br><br>1. Gap release of activity from a fuel handling accident and heavy load drops on spent fuel not exceeding EPA PAGs.<br>2. A fire involving resin and gamma radiation due a loss of spent fuel pool level not exceeding EPA PAGs.<br>3. A bounding scenario where the fuel is totally uncovered and no natural circulation flow path exists. The staff calculated that it would take ~14 hours to heat up to 900 degrees C.<br><br>The staff concluded that the postulated doses to the general public from any reasonably conceivable accident would not exceed EPA PAGs and, for the bounding accident, the length of time available gives confidence that mitigative actions and, if necessary, offsite measures for the public could be taken without preplanning. |
| Zion | 2/13/98 | 8/31/99 | The staff concluded that there were no design basis accidents or other credible events that would result in a radiological dose beyond the exclusion area boundary that would exceed EPA PAGs.<br><br>For a bounding scenario where the fuel is totally uncovered, the decay heat would not heat up higher than 482 degrees C; therefore the cladding would stay intact. |

## Attachment 3
## Industry Decommissioning Commitments and Staff Decommissioning Assumptions

### Industry Decommissioning Commitments (IDCs)

IDC #1 Cask drop analyses will be performed or single failure-proof cranes will be in use for handling of heavy loads (i.e., phase II of NUREG-0612 will be implemented).

IDC #2 Procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event.

IDC #3 Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.

IDC #4 An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The plan would principally identify organizations or suppliers where offsite resources could be obtained in a timely manner.

IDC #5 Spent fuel pool instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for spent fuel pool temperature, water level, and area radiation levels.

IDC #6 Spent fuel pool seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self-limiting to leakage or otherwise engineered so that drainage cannot occur.

IDC #7 Procedures or administrative controls to reduce the likelihood of rapid draindown events will include: (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.

IDC #8 An onsite restoration plan will be in place to provide repair of the spent fuel pool cooling systems or to provide access for makeup water to the spent fuel pool. The plan will provide for remote alignment of the makeup source to the spent fuel pool without requiring entry to the refuel floor.

IDC #9 Procedures will be in place to control spent fuel pool operations that have the potential to rapidly decrease spent fuel pool inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.

IDC #10 Routine testing of the alternative fuel pool makeup system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available, if needed.

**Attachment 3**
**Industry Decommissioning Commitments and Staff Decommissioning Assumptions**

## Staff Decommissioning Assumptions (SDAs)

SDA #1 Licensee's SFP cooling design will be at least as capable as that assumed in the risk assessment, including instrumentation. Licensees will have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP:

Makeup pump: 20-30 gallons per minute (gpm)
Firewater pump: 100-200 gpm
Fire engine: 100-250 gpm (100 gpm, for 1 1/2-in hose, 250 gpm for 2 1/2-in. hose)

SDA #2 Walk-downs of SFP systems will be performed at least once per shift by the operators. Procedures will be developed for and employed by the operators to provide guidance on the capability and availability of onsite and offsite inventory makeup sources and time available to initiate these sources for various loss of cooling or inventory events.

SDA #3 Control room instrumentation that monitors SFP temperature and water level will directly measure the parameters involved. Level instrumentation will provide alarms at levels associated with calling in offsite resources and with declaring an emergency.

SDA #4 Licensee determines that there are no drain paths in the SFP that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level.

SDA #5 Load drop consequence analyses will be performed for facilities with non-single failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining would be sufficient to demonstrate that there is high confidence in the facilities ability to withstand a heavy load drop.

SDA #6 Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 2B to NUREG-1738. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in NUREG-1738 ($<1 \times 10^{-5}$ per year including non-seismic events).

SDA #7 Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high-density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.

# EXHIBIT 11

**STATE OF WASHINGTON**
**DEPARTMENT OF ECOLOGY**

| | | |
|---|---|---|
| IN THE MATTER OF AN<br>ADMINISTRATIVE ORDER<br>AGAINST | )<br>)<br>) | ADMINISTRATIVE ORDER<br>DOCKET #10156 |

United States Department of Energy
Mr. Kevin Smith, Program Manager
Office of River Protection
PO Box 450, MSIN: H6-60
Richland, Washington 99352

Washington River Protection Solutions
Mr. L. David Olson, President & Project Manager
PO Box 850, MSIN: H6-04
Richland, Washington 99352

| | |
|---|---|
| **Order Docket #** | 10156 |
| **Site Location** | The Hanford Site within Benton, Franklin, and Grant Counties of Washington |
| **EPA ID** | #WA 7890008967 |

The Washington State Department of Ecology (Ecology) issues this Administrative Order (Order) requiring the U.S. Department of Energy (USDOE) and Washington River Protection Solutions (WRPS) to comply with:

- Chapter 70.105 Revised Code of Washington (RCW) Hazardous Waste Management Act.
- Chapter 173-303 Washington Administrative Code (WAC) Dangerous Waste Regulations.
- Hanford Facility Dangerous Waste Permit, No. WAD WA7890008967 (Permit).

## AUTHORITY

Ecology is authorized under RCW 70.105.095 to issue an administrative order requiring compliance upon determining that a person has violated, or is about to violate, any provision of Chapter 70.105 RCW.

RCW 70.105.130 authorizes Ecology to implement the federal Resource Conservation and Recovery Act (RCRA), and establish a permit system for owners or operators of facilities that treat, store, or dispose of dangerous waste. The permit system is established in the Dangerous Waste Regulations, Chapter 173-303 WAC.

Ecology issued Permit No. WAD WA7890008967 (Permit) for USDOE's Hanford Dangerous Waste Facility (Facility), effective August 1994. Revision 8c of the Permit currently applies to the operation of and corrective actions taken, or to be taken, at, this Facility.

Pursuant to Part I.A of the Permit, Revision 8c, the standards used to evaluate compliance for this enforcement are the interim status facility standards in WAC 173-303-400 and the regulations incorporated into the interim status standards by reference.

## FACTUAL FINDINGS

Ecology's determination that a violation has occurred is based on the following facts:

1. Tank 241-AY-102 is one of two one-million gallon tanks in the 241-AY Tank Farm (AY Farm) located in the southeast portion of the 200 East Area of the Hanford Dangerous Waste Facility. The 241-AY-102 system includes:

   - Primary tank and secondary tank structure
   - Concrete shell, insulating pad (refractory), and foundation
   - Central pump pit
   - Sluice pits
   - Annulus pump pit
   - Leak detection pit (and well)
   - Air lift circulators
   - Monitoring and alarm systems

   The primary steel tank rests inside the secondary steel tank and is supported by the refractory on the floor of the secondary tank. An annular space of 2.5 feet is formed between the primary tank and secondary tank.

2. In August 2012, an accumulation of material was discovered at two locations on the floor of the 241-AY-102 annulus that separates the primary tank from the secondary tank.

   The accumulation of material was discovered during a routine video inspection. None of this material was present during the last visual inspection of the annulus, taken in 2006 - 2007. USDOE and WRPS conducted further investigation and sampling, and determined that the accumulated material was leaking from the primary tank.

3. On October 22, 2012, USDOE notified Ecology that Tank 241-AY-102 was leaking waste into the tank's secondary containment.

4. Hazardous and highly radioactive waste material cascaded from refractory slots to the floor of the annulus in two locations – near Riser 90 and near Riser 83. The flow near Riser 90 has shown no changes since the notification. The flow near Riser 83 shows a continuing leak. As of November 15, 2012, the amount of hazardous and radioactive waste material that has leaked from both areas was approximately 520 gallons.

5. On March 5, 2014, USDOE notified Ecology that a third leak had been discovered from Riser 77. The volume of this leak is unknown at this time.

6. Through a series of meetings and other interactions from October 2012 through the date of this Order, Ecology has given USDOE and WRPS opportunities to voluntarily comply with applicable regulations.

   During this period, Ecology stated numerous times, both orally and in written form, that the leak response requirements at 40 CFR 265.196 [incorporated by reference into interim status standards at WAC 173-303-400(3)][1] apply and must be complied with. In particular, 40 CFR 265.196(b) requires removal of waste from the primary tank and secondary containment of a leaking tank system.

7. By email on October 23, 2012, Ecology told USDOE and WRPS that waste removal must begin immediately, and requested a detailed schedule for completing such removal. USDOE was told to immediately notify Ecology if it did not intend to comply with the requirements.

8. By email on December 3, 2012, Ecology again reminded USDOE and WRPS of the requirements to immediately pump Tank 241-AY-102 or provide a schedule and justification for completing this at the earliest practicable time.

9. On January 15, 2013, WRPS, along with USDOE, presented three options for coming into compliance with 40 CFR 265.196:

   (1) Closure pursuant to the requirements of 40 CFR 265.196(e).

   (2) Repair and recertification pursuant to the requirements of 40 CFR 265.196(e) and (f).

   (3) Obtaining a secondary containment variance pursuant to the provisions for this at 40 CFR 265.193 (g) and (h).

   Ecology informed WRPS and USDOE that these options did not meet the tank leak response requirements in 40 CFR 265.196. In particular, none of these options addressed the requirement to, within 24 hours, or if that is demonstrably not possible, at the earliest practicable time, remove as much of the waste as necessary to allow for tank system inspection.

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[1] For brevity purposes, for the remainder of the Order Ecology will only cite to 40 Code of Federal Regulations (CFR) 265.196. In all cases, however, the citation is to the federal regulation as it is incorporated by reference under WAC 173-303-400(3).

10. In multiple meetings and conversations since December 2012, Ecology requested that USDOE provide a response to the regulatory requirements and a plan to remove the waste from Tank 241-AY-102. Ecology reviewed and provided comments to USDOE on multiple versions of a draft letter intended to provide this information.

11. On May 6, 2013, USDOE provided a letter to Ecology that:

    (1) Provided a regulatory basis for not pumping the tank within 24 hours.

    (2) Indicated that the tank was not isolated from waste additions.

    (3) Indicated that the ability of the secondary tank to maintain integrity, once waste entered it, was still under evaluation.

    (4) Committed to provide to Ecology a pumping plan specific to Tank 241-AY-102 by June 14, 2013.

12. On May 24, 2013, Ecology issued a letter to USDOE and WRPS documenting its expectations for the June 14, 2013, pumping plan submittal. The letter conveyed the following expectations:

    - The pumping plan must provide a schedule for removing waste from the primary tank. [40 CFR 265.196(b)(1)]
    - The pumping plan must provide a schedule for removing waste from the secondary containment, demonstrating that such removal is in as timely a manner as is possible to prevent harm to human health and the environment. [40 CFR 265.196(b)(2)]
    - The pumping plan must provide a schedule for isolating the 241-AY-02A pit, which could provide a path to allow waste into Tank 241-AY-102.
    - The pumping plan must provide a schedule to revise the January 2006 evaluation of the integrity of the secondary containment.
    - The pumping plan must document technical challenges that may affect the schedule, separate from limitations on funding. Funding may not be a factor in determining "earliest practicable time" or "as timely as possible." [40 CFR 265.196(b)]
    - The pumping plan must document readiness to pump, within a specific and reasonable timeframe, from both the primary tank and secondary containment, if the leak worsens.
    - An earlier-prepared Emergency Pumping Guide must be immediately revised because it did not fulfill the goal of allowing pumping of a double-shelled tank (DST) within 10 days if a leak occurred, determined through previous compliance actions to address 40 CFR 265.196(b).

      Ecology conveyed this expectation because USDOE had documented its belief that the earlier Emergency Pumping Guide does not apply to a leak from the bottom of a tank.

13. On June 14, 2013, USDOE delivered the "241-AY-102 Pumping Plan," RPP-PLAN-55220, Rev. A, (Pumping Plan) to Ecology. Through this Pumping Plan and its accompanying letter, USDOE:

    - Declined to remove any waste from the primary tank unless conditions change, stating that "removal of waste from the primary tank is not practicable, nor is it necessary to prevent release to the environment."
    - Laid out a schedule of approximately 19 months for "planning, procurement and installation of the out-of-tank equipment that will be needed to allow for pumping of the solids in the primary tank."
    - Declined to schedule installation of in-tank pumping equipment necessary for solids removal during the 19 months of planning, procurement, and installation of equipment.
    - Indicated that waste removal, if initiated, would take 14 months to complete, after which the tank would be evaluated for repair or closure.
    - Expressly assumed that "the secondary containment will remain intact until waste from tank AY-102 can be removed and the 'repair or close' decision made."
    - Commits to completing a study on the structural integrity of secondary containment by April 2014.

14. The impact of the waste in the Tank 241-AY-102 annulus on the integrity of the secondary liner is unknown at this time.

15. USDOE has taken no action to mitigate the leak into the secondary containment. As of the date of this Order, USDOE has taken no action to prevent the flow of dangerous waste into Tank 241-AY-102 or stop the flow of waste into its secondary containment.

16. The Defense Nuclear Facilities Safety Board (DNFSB) reviewed USDOE report RPP-RPT-53901, Rev. 2, "Management of Supernatant Level in Tank 241-AY-102." On October 24, 2013, the DNFSB released a staff issue report on the subject of "Integrity Implications of Decanting Liquid from Hanford Tank 241-AY-102."

    The report recommends:

    - Continued visual inspection of the tank annulus and close monitoring for variations in the waste temperature.
    - Monitor for signs of increased leakage and blockage of the insulating refractory slots that distribute cooling air to the tank bottom.
    - Develop a more rigorous multi-dimensional, transient thermal analysis model to aid in understanding the safety significance of any observed changes in tank conditions subsequent to decanting.

17. On January 9, 2014, Ecology issued a letter to USDOE and WRPS with comments on the 241-AY-102 Pumping Plan, Rev. A (RPP-PLAN-55220). Ecology found the Pumping Plan unacceptable because the plan expressly declined to meet USDOE's and WRPS's regulatory obligation to remove the waste from Tank AY-102 at the earliest practicable time. Ecology's letter requested a workable plan for pumping waste from Tank AY-102 no later than February 15, 2014

18. On February 4, 2014, USDOE issued a letter to Ecology asking for an extension for submittal of the revised 241-AY-102 Pumping Plan to March 7, 2014

19. On February 11, 2014, Ecology issued a letter to USDOE approving the extension to March 7, 2014.

20. On March 7 USDOE submitted Rev C of the Revised 241-AY-102 Pumping Plan (revised Pumping Plan). The plan announces that it "has been revised to proceed with the planning, engineering and design, procurement, and installation of out of tank equipment." However, it does not contain a plan for conducting these activities. Its only schedule is an estimated timeframe of approximately two years for conducting only the preparatory activities.

    The revised Pumping Plan indicates this estimated timeframe may be subject to change for various reasons. The plan does not attempt to show that this two-year timeframe satisfies the requirement of "earliest practicable time." It does not provide any plan or schedule for actually removing the waste from Tank 241-AY-102.

## DETERMINATION OF VIOLATIONS

Ecology has determined that the following violations have occurred based on the facts provided above.

### Violation 1 - Failure to stop the flow of hazardous waste into secondary containment

*40 CFR 265.196(a) requires the owner or operator of the tank to immediately stop the flow of hazardous waste into the secondary containment system.*

As of the date of this Order, USDOE and WRPS have not stopped the flow of waste into the secondary containment of 241-AY-102.

**Violation 2 - Failure to inspect the tank to determine the cause of the release**

*40 CFR 265.196(a) requires the owner or operator of the tank to inspect the tank to determine the cause of the release.*

As of the date of this Order, USDOE and WRPS have not inspected the tank to determine the cause of the release. USDOE states in the revised Pumping Plan that Tank 241-AY-102 will have to be emptied to determine the cause of the release. USDOE has not emptied the tank and has submitted a plan according to which waste removal will not be authorized, nor a removal schedule determined, before March 4, 2016. The revised plan does not demonstrate that an initial pumping date sometime after March 4, 2016 is the earliest practicable time to begin waste removal.

**Violation 3 - Failure to remove, at the earliest practicable time, as much of the waste as is necessary to prevent further release of hazardous waste to the environment and to allow inspection and repair of the tank to be performed.**

*Where the release is from the tank system, as it is here, 40 CFR 265.196 (b) provides that "the owner or operator must, within 24 hours after detection of the leak or, if the owner or operator demonstrates that that is not possible, at the earliest practicable time remove as much of the waste as is necessary to prevent further release of hazardous waste to the environment and to allow inspection and repair of the tank system to be performed."*

As of the date of this Order, USDOE and WRPS have failed to remove, or take any actions to begin removing, as much of the waste as is necessary to prevent further release to the environment and to allow for inspection and repair of the tank system to be performed.

USDOE states in its revised Pumping Plan that removing the contents of the tank will not be authorized before March 4, 2016. USDOE has not demonstrated that March 4, 2016, or later would be the "earliest practicable time" to begin removing the waste.

**Violation 4 - Failure to remove all released materials from the secondary containment system within 24 hours or in as timely a manner as is possible to prevent harm to human health and the environment**

*40 CFR 40 CFR 265.196 (b)(2) requires that, if the release was to a secondary containment system, all released materials must be removed within 24 hours or in as timely a manner as is possible to prevent harm to human health and the environment.*

As of the date of this Order, USDOE and WRPS have failed to remove any of the released materials from the secondary containment. The revised plan indicates that the released materials will be removed only after waste is removed from the primary tank.

## ORDER TO COMPLY

Based on the factual findings and the determinations of violations, as stated above, IT IS ORDERED THAT USDOE and WRPS take the actions described below.

Immediately upon receipt of this Order and continuously thereafter USDOE and WRPS must:

1. Provide to Ecology, upon publication, the results of any modeling that USDOE or WRPS conducts in accordance with recommendations of the DNFSB staff report, "Integrity Implications of Decanting Liquid from Hanford Tank 241-AY-102" (October 24, 2013).

2. Complete isolation of Tank 241-AY-102 by August 15, 2014.

3. After the 241-AY-02A pump pit has been isolated, and no later than September 1, 2014, begin pumping the supernatant from Tank 241-AY-102. Remove all supernatant, except as necessary to maintain the minimum height of supernatant above the maximum solids level prescribed in RPP-RPT-53901 (prescribing 96 inches above solids level), or as prescribed in other USDOE documents regulating safety in Tank 241-AY-102.

4. Complete installation of sludge removal equipment and initiate waste removal in Tank 241-AY-102 no later than December 1, 2015. This will include all activities that USDOE will need to complete for authorization to initiate and complete all waste transfers.

5. Complete waste removal to a level sufficient for inspection to determine the cause of the leaks, no later than December 1, 2016.

6. Immediately inform Ecology of any safety issues that arise after pumping has begun and provide a detailed description of the specific safety issue. If the solution to an immediate concern is to cease pumping, provide a recovery plan within 30 days. The recovery plan must include a schedule for correcting and restarting pumping at the earliest practicable time.

7. Within 60 days of the effective date of this Order, submit to Ecology for approval:

    a. Monitoring plans for annulus inspection, waste temperature monitoring and annulus ventilation monitoring including a schedule for calibration of the continuous air monitor (CAM) and Enraf-Nonius Series 854 (ENRAF). The monitoring plans must provide clear, immediate actions for maintaining annulus ventilation.

    b. A contingency plan for safely managing any worsening conditions indicated by inspections and monitoring. Such indications include suspected increased leak rate or blockage on the ventilation channels causing increases in waste temperatures.

    Any other new issues not identified in the contingency plan such as those that arise as a result of construction or waste transfer activities, must be identified and evaluated, with a recovery plan and schedule provided to Ecology within 30 days.

8. Within 90 days of the effective date of this Order, submit a report that evaluates the integrity of the secondary containment system including, but not limited to, the impacts of the waste that is currently in the annulus.

9. Within 120 days of the effective date of this Order, submit a detailed waste retrieval work plan to Ecology for removing the remaining waste from Tank 241-AY-102. The waste retrieval work plan shall include, but is not limited to, detailed descriptions of:

   a. The engineering design and the steps taken to procure equipment, including those steps already undertaken, with a schedule for the procurement of each piece of equipment, showing that these activities either have been or will be completed at the earliest practicable time.

   b. The steps necessary for installation of all needed out-of-tank equipment and in-tank equipment for removing the waste from Tank 241-AY-102.

   c. The number and schedule of 242-A Evaporator runs, including support activities needed.

   d. The schedule for installation and start-up of equipment needed to support transfers to other DSTs.

10. Officially submit all supporting documentation that justifies the schedule for the above requirements.

11. To address the potential leak to the environment, sample the liquid from the Tank 241-AY-102 annulus leak detection pit monthly, starting within five days of the effective date of this Order. At a minimum, using inductively coupled plasma/mass spectrometry (ICP/MS), analyze this sample for metals, radionuclides, and pH, and report the results to Ecology within 15 days of taking the sample.

12. Conduct monthly video inspections of the entire annulus and weekly video inspections on the current leaks and weekly video inspections of any future leaks into the annulus.

13. Provide Ecology with monthly reports on the results of the visual and video annulus inspections, annulus ventilation performance and status, CAM readings, ENRAF readings, CAM and ENRAF calibration results, sample analysis results, waste heat monitoring results, including any interpretations and conclusions based on the results.

14. Officially submit to Ecology, within 10 working days of the effective date of the Order, copies of:

   a. All documents listed in the revised Pumping Plan, Attachment A, that were not previously officially submitted to Ecology

   b. All Technical Safety Requirements and all Safety Basis evaluations used to determine the requirements to control flammable gas levels and impacts to operational limits for waste storage (OSD-T-151-00007), as referenced in the revised Pumping Plan, Section 1.1, that were not previously officially submitted to Ecology

## EFFECTIVE DATE

This Order is to be considered effective 30 days from the day of issuance.

## ELIGIBILITY FOR PAPERWORK VIOLATION WAIVER AND OPPORTUNITY TO CORRECT

Under RCW 34.05.110, small businesses are eligible for a waiver of a first-time paperwork violation and an opportunity to correct other violations.

Ecology has determined the requirements of RCW 34.05.110 do not apply to the violation(s) described in this Order because you are not a small business as defined in RCW 34.05.110 (9).

## FAILURE TO COMPLY WITH THIS ORDER

Failure to comply with this Order may result in the issuance of civil penalties or other actions, whether administrative or judicial, to enforce the terms of this Order.

## YOUR RIGHT TO APPEAL

You have a right to appeal this Order to the Pollution Control Hearing Board (PCHB) within 30 days of the date of receipt of this Order. The appeal process is governed by Chapter 43.21B RCW and Chapter 371-08 WAC. "Date of receipt" is defined in RCW 43.21B.001(2).

To appeal you must do both of the following within 30 days of the date of receipt of this Order:

- File your appeal and a copy of this Order with the PCHB (see addresses below). Filing means actual receipt by the PCHB during regular business hours.
- Serve a copy of your appeal and this Order on Ecology in paper form - by mail or in person. (See addresses below.) E-mail is not accepted.

You must also comply with other applicable requirements in Chapter 43.21B RCW and Chapter 371-08 WAC.

- **ADDRESS AND LOCATION INFORMATION**

| Street Addresses | Mailing Addresses |
|---|---|
| **Department of Ecology**<br>Attn: Appeals Processing Desk<br>300 Desmond Drive SE<br>Lacey, Washington 98503 | **Department of Ecology**<br>Attn: Appeals Processing Desk<br>PO Box 47608<br>Olympia WA 98504-7608 |
| **Pollution Control Hearings Board**<br>1111 Israel Road SW, Suite 301<br>Tumwater, Washington 98501 | **Pollution Control Hearings Board**<br>PO Box 40903<br>Olympia WA 98504-0903 |

## CONTACT INFORMATION

Please direct all questions about this Order to:

Nina M. Menard, Acting Section Manager
Department of Ecology
Nuclear Waste Program
3100 Port of Benton Boulevard
Richland, Washington 99354

(509) 372-7972

Nina.Menard@ecy.wa.gov

## MORE INFORMATION

- **Pollution Control Hearings Board**
  www.eho.wa.gov/Boards_PCHB.aspx
- **Chapter 43.21B RCW - Environmental and Land Use Hearings Office – Pollution Control Hearings Board**
  http://apps.leg.wa.gov/RCW/default.aspx?cite=43.21B
- **Chapter 371-08 WAC – Practice and Procedure**
  http://apps.leg.wa.gov/wac/default.aspx?cite=371-08
- **Chapter 34.05 RCW – Administrative Procedure Act**
  http://apps.leg.wa.gov/RCW/default.aspx?cite=34.05
- **Chapter 70.105 RCW – Hazardous Waste Management**
  http://apps.leg.wa.gov/rcw/default.aspx?cite=70.105
- **Chapter 173-303 WAC – Dangerous Waste Regulations**
  http://apps.leg.wa.gov/wac/default.aspx?cite=173-303

## SIGNATURE

Maia D. Bellon

Maia D. Bellon
Director

3/21/14
Date

# EXHIBIT 12

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# Reducing the Risks of High-Level Radioactive Wastes at Hanford

Robert Alvarez

Senior Scholar, Institute for Policy Studies, Washington, DC, USA. This work was done in collaboration with the Government Accountability Project.

High-level radioactive wastes resulting from plutonium production at the U.S. Department of Energy's (DOE) Hanford site in Washington State are among the largest and most dangerous byproducts of the nuclear arms race. The Energy department announced plans in 2002 to terminate its environmental mission at Hanford and all other DOE sites over the next 30 years. During this time, DOE intends to dispose of approximately 90 percent of Hanford's high-level wastes onsite, process the remainder into glass for geological disposal, and permanently close 177 large tanks, and related infrastructure. Central to the department's goal at Hanford is to speed up, perhaps, the most expensive, complex, and risky environmental project in the United States. Estimated life-cycle costs for processing Hanford's wastes are between $41.6 and $56.9 billion. No country has processed anything quite like Hanford's large and complex brew of wastes.

## PROCESSING RISKS

The accident consequences at Hanford's Waste Treatment Plant are comparable to those accidents at a large nuclear reactor. During design and construction of a nuclear facility, DOE is required by regulation to estimate the frequency of unmitigated risks of major nuclear accidents, which do not account for preventive features that would lessen the consequences of an accident. This approach defines the "safety envelope" contained in documented nuclear safety analyses, required for the regulation of design, constructions, and operation of a nuclear facility. Its purposes are to encourage higher margins of safety and to envelop uncertainties inherent with first-of-a-kind, hazardous operations.

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Address correspondence to Robert Alvarez, Senior Scholar, Institute of Policy Studies, 733 15th St. NW, Suite 1020, Washington, DC, 20005, USA. E-mail: kitbob@erols.com

After three and a half years of involvement at Hanford, NRC found in 2001 that DOE contractors consistently downplayed the severity of potential accidents. NRC estimated that the overall unmitigated risk of major radiological and chemical accidents at Hanford's high-level waste operations was 2.4E-2/yr, translating into a 50-50 chance over an estimated 28 years of operation of the facility. According to NRC more than two-dozen significant safety issues and 50 specific topics remained unresolved.

Existing engineering controls and administrative methods can reduce accident risks at the Hanford Waste Treatment Plant to acceptable levels, with the possible exception of glass melters, designed to mix radioactive wastes with molten glass. They will be the largest in the world and pose potentially the most severe accident consequences. NRC found that further analysis was required to determine if melter risks could be reduced to levels acceptable for reactor accidents. But, NRC warned that "few tests appear to be planned to verify safety parameters prior to construction."

DOE's experience with glass melters does not inspire confidence. Since 1991 there have been at least eight melter-related accidents and failures at DOE sites, including two steam explosions.

Storage problems stemming from Cold War practices add significant risks to waste processing. More than a third of Hanford's tanks have leaked approximately one million gallons, contaminating groundwater that eventually enters the Columbia River. The structural integrity of dozens of aged tanks "represent immediate concerns," says the Nuclear Regulatory Commission (NRC). As a result of an early decision to neutralize acidic reprocessing wastes, all of Hanford's tanks generate potentially flammable and explosive gases from radiolysis. This problem is exacerbated by hundreds of added chemicals.

One of Hanford's most troublesome tanks, SY 101, was found in October 2003, after it was declared safe, to have a sufficient amount of retained gas to reach 100 percent of the lower flammable limit for hydrogen. As wastes are retrieved and processed, the risks of fires and explosions can increase, and these will be a concern throughout the project life. Current estimates indicate that hydrogen build-up in pipes at the Waste Treatment Plant could be tens to hundreds of times greater than assumed.

"On many occasions, there was an implication that regulatory reviews were not allowed to impact cost and schedule," NRC concluded. Since 2001, when NRC involvement ended at the Hanford Waste Treatment Plant, DOE appears to not have heeded NRC's numerous concerns. Instead, programmatic demands to reduce cost and save time, have led to relaxed safety requirements, higher construction costs, and increased worker exposures and injuries. A recent DOE inquiry found that construction workers expressed fear of retaliation, particularly job loss, for reporting safety, medical, and labor relations issues. As a result of these growing problems, construction has been curtailed and design work has to be revisited, causing further delays.

## DISPOSAL RISKS

Stating that the planned Yucca Mountain geological repository "does not have the space," DOE seeks to greatly expand on-site burial of defense high-level wastes at several sites. The underlying bases for this policy were established in 1985. They assume that a high-level waste canister contains the radiological equivalent of 0.5 metric tons heavy metal of spent power reactor fuel, and that the preponderance of wastes in Hanford's tanks would be effectively abandoned. Current data from 1,500 high-level waste packages produced at the Savannah River Site indicate that canisters contain less than 10 percent of the predicted radioactivity. DOE is also prohibited by a federal environmental compliance agreement from abandoning Hanford tanks. Given these circumstances, DOE's policy to further reduce high-level waste canister production will to lead to the on-site disposal of substantially larger amounts of radionuclides.

Before DOE initiated an accelerated cleanup plan in 2002, at least 98 percent of the total radioactivity was to be removed from soluble wastes at Hanford, under a 1997 agreement with the NRC staff prior to on-site disposal, as incidental wastes. Instead, DOE intends to bury wastes on-site from dozens of tanks without radionuclide separation, as well as undetermined amounts of tank residuals, and failed processing equipment containing high-level wastes. As a result, at least 35 megacuries of radioactivity could be disposed on-site at Hanford—more than twice the amount agreed to in 1997 by the NRC staff.

Prior to 2004, the NRC determined what constitutes high-level wastes at DOE sites for geological disposal. Last year the US Congress authorized the DOE to self-regulate high-level waste disposal, with NRC consultation. However, the Hanford site was excluded from this provision. NRC has exercised sits authority through staff-level consensus agreements. NRC's passive approach has resulted in DOE disposal actions, which for all practical purposes are irreversible. In the case of Hanford, it appears that DOE can ignore an agreement with NRC staff, without regulatory consequence.

The National Research Council recently concluded that knowledge of the fate and transport of tank wastes into the Columbia River is tenuous, at best; and that premature failure of environmental barriers is likely. Current estimated disposal of iodine-129 (17 million year half-life) from processing wastes, appear to violate DOE's waste performance requirements and could contaminate groundwater in excess of EPA drinking water limits for thousands of years.

The impacts of past operations and additional onsite disposal have not been assessed on the Hanford Reach, the last free flowing stretch of the Columbia River, which runs though the site. Higher priority is being given to the transfer of more than 87 percent of the Hanford site over the next eight years to the Interior Department's Hanford Reach National Monument. The likelihood of thousands of people visiting the Monument has not been factored into DOE's

nuclear accident scenarios, or disposal decisions. Of particular concern is the high vulnerability to environmental contaminants of thousands of tribal people living near Hanford.

This was underscored in 2002 by the U.S. Environmental Protection Agency which found that fish near the site have the highest contaminant concentrations in the Main Stem Columbia River Basin. EPA estimated that lifetime fatal cancer risks from fish consumption to tribal people are as high as 1 in 50. Usually, EPA takes regulatory action when contaminant risks exceed 1 in 10,000 to 1 in 1,000,000. Around the time the EPA study was released, DOE set a radiation standard to protect fish from Hanford's radionuclides that would result in doses to tribal people several hundred times greater than allowed by the Environmental Protection Agency.

## PROJECT MANAGEMENT ISSUES

Since the Hanford waste treatment plant is a first-of-a-kind endeavor, safety and operability of this project is highly dependent on knowledge of physical and chemical properties of the wastes. However, the National Research Council finds that Hanford waste data "is of little value in designing chemical remediation processing." In light of these uncertainties, worldwide high-level waste vitrification experience encourages extraordinary caution be exercised at Hanford. But DOE has raised the stakes by deciding to forego a pilot plant using actual Hanford wastes and to concurrently design and construct a full-scale facility.

Over the past 20 years, less than five percent of all defense high-level wastes have been processed, while incurring soaring costs, projected to exceed $100 billion. DOE's inability to manage these projects is a major factor behind these difficulties. For instance, a 20-year failure to pretreat soluble wastes at the Savannah River Site has resulted in a loss of $500 million, with projected costs of $1.8 billion. The U.S. General Accounting Office attributes this problem to a management culture based on an "undocumented policy of blind faith in its contractors' performance." Growing Congressional concern has resulted in recent reports by the National Research Council, which found:

- Environmental projects suffer from major delays and are about 50 percent more expensive than comparable federal and private-sector projects;
- Up-front project planning is inadequate;
- There is no consistent system for evaluating project risks; and
- DOE is not in control of many of its projects.

Capital costs for the Hanford vitrification plant are a relatively small portion of the total life-cycle costs for the project. The failure to address critical

uncertainties in the design and construction of the plant could significantly impact processing and disposal costs and the overall success of this endeavor. DOE's policy to put concurrent design and construction on a "fast track" has led to costly and time-consuming mistakes.

The most significant shortcoming to date has been DOE and the contractor's failure to heed warnings by the Energy department's Defense Nuclear Facility Safety Board in 2003 to increase hardening against earthquakes in a seismically active region similar to that as California. As a result, in March 2005, DOE had to suspend construction work on facilities that would process the preponderance of the wastes, in order to double the seismic design standard.

## RECOMMENDATIONS

To reduce the risks of Hanford's high-level wastes this article makes the following recommendations.

- The Nuclear Regulatory Commission should be authorized to regulate the design and construction of Hanford's waste processing operations and certify the safety of storage tanks.
- Risk-based criteria identified by the DOE that would allow for the geological disposal of all DOE HLW canisters should be adopted.
- More restrictive limits for on-site disposal of tank wastes should be imposed on the permanent on-site disposal of high-level tank wastes. These limits should be developed with affected states, Indian tribes and public stakeholders. This should be done under existing law, through formal rulemaking by the NRC.
- The Energy department should build pilot operations for high-level waste pretreatment, feed preparation, and melters using actual Hanford wastes.
- DOE should strengthen its oversight of this project by establishing a full-time Hanford high-level waste processing project management group, reporting to the Assistant Secretary for Environmental Management.

The costs, complexity and risks of the Hanford high-level waste project plant rival those of the U.S Space Shuttle program, but have far greater potential consequences to the human environment. Yet it remains for the most part, an expensive curiosity in national policy deliberations. Given the stakes involved, the price of continued obscurity of this legacy of the nuclear arms race may prove to be incalculable.

## HANFORD HISTORY

In January 1943, just weeks after the world's first self-sustaining nuclear chain reaction took place at the University of Chicago, the Hanford site in the steppe shrub desert of Southeastern Washington was selected to make plutonium for the first atomic weapons. Its relative isolation and close proximity to the large water and electrical supplies from the Columbia River made the 560-square-mile site a seemingly ideal location. Over the next 44 years, until U.S. Energy Secretary John Herrington announced that the nation was "awash in plutonium,"[1] Hanford's nine reactors had produced 67.4 metric tons of this fissile material.[2]

As Cold War memories fade, the sobering aftermath of the nuclear arms race is no more apparent than at Hanford, where the nation's most hazardous byproducts of nuclear weapons production are stored. With nearly 60 percent of the nation's defense high-level radioactive waste,[3] Hanford's legacy is in a league unto itself in terms of magnitude and risk.

The United States started to come to terms with this problem when the Congress established a process to dispose of geologically defense and civilian high-level radioactive waste in the 1982 Nuclear Waste Policy Act. The following year, borosilicate glass was selected by the U.S. Department of Energy (DOE) as the preferred disposal form for defense high-level wastes. Vitrified wastes from Hanford, and four other sites,[4] would then be disposed along with DOE and commercial spent power reactor fuel in the same repository.

After more than 20 years of fits, starts, and soaring costs, less than five percent of the nation's defense high-level wastes have been processed.[5] At Hanford, DOE is still in the stages of design and construction. The success of this unprecedented endeavor, estimated to cost between $41.6 and $56.9 billion,[6] depends largely on the resolution of three key questions.

1. Can the processing of Hanford's high-level wastes be done safely?
2. Can DOE "fast track" a full-scale, first-of-a-kind operation, without significant technological failures?
3. Will the shallow on-site disposal of radioactive wastes from Hanford's tanks ensure adequate protection of health and natural resources from the present to thousands of years from now?

## HANFORD'S HIGH-LEVEL WASTES

High-level wastes were generated by dissolution of 119,271 MTU (metric tons uranium) of spent reactor fuel[7] and the subsequent solvent extraction of plutonium and other materials.[8] Because of their highly intense radioactivity they must be handled remotely in heavily shielded structures. Until recently, their

long-term hazards required that they be disposed so as to protect the human environment for up to 10,000 years.[9] However, this standard was struck down by the United States Court of Appeal,[10] citing the National Research Council's finding that peak radiation doses "might occur tens to hundreds of thousands of years into the future."[11]

In making these wastes at Hanford, five chemical processes were utilized.[12] After treatment and subsequent radioactive decay,[13] the Hanford high-level wastes currently contain approximately 194 megacuries[14] in 54 million gallons (204,000 cubic meters) stored in large underground tanks.[15] (See Table 1). From a time perspective, radionuclides in the tanks pose potentially significant risks to health and natural resources for 300 to more than 200,000 years.[16]

More than 96 percent of total radioactivity in the tank wastes comes from cesium-137 and strontium-90 (half-lives of 30 and 29 years respectively). These high levels of radiostrontium and radiocesium pose safety concerns because of decay heat build-up during storage, retrieval, and processing.[17] Hanford's wastes also have substantial amounts of long-lived fission products and transuranics. The amount of technetium-99 produced at Hanford (200,000 year half-life) is nearly nine times more than that released from all world-wide atmospheric nuclear weapons tests.[18] Because of its rapid mobility, Tc-99 can

**Table 1:** Hanford high-level waste inventory.

| Radionuclides (Ci) | | Radionuclides (cont) | | Other analytes (Kg) | |
|---|---|---|---|---|---|
| 3H | 1.04E+04 | 152Eu | 1.71E+03 | F | 1.08E+06 |
| 90Y | 4.99E+07 | 14C | 3.01E+03 | Al | 8.05E+06 |
| 90Sr | 4.99E+07 | 137Cs | 4.62E+07 | Fe | 1.25E+06 |
| 60Co | 8.08E+03 | 137mBa | 4.37E+07 | La | 3.69E+04 |
| 234U | 2.21E+02 | 129I | 4.79E+01 | Pb | 7.84E+04 |
| 106Ru | 1.02E+03 | 227Ac | 1.30E+02 | Mn | 1.65E+05 |
| 134Cs | 1.82E+04 | 243Am | 1.52E+01 | Hg | 1.83E+03 |
| 233U | 5.08E+02 | 239Pu | 6.88E+04 | Ni | 1.16E+05 |
| 244Cm | 2.88E+02 | 235U | 9.14E+00 | K | 9.18E+05 |
| 238Pu | 4.23E+03 | 228Ra | 6.24E+01 | Si | 8.01E+05 |
| 63Ni | 1.28E+05 | 242Cm | 1.44E+02 | Na | 4.80E+07 |
| 242Pu | 8.29E-01 | 154Eu | 1.02E+05 | Sr | 3.96E+04 |
| 226Ra | 2.38E+02 | 229Th | 2.58E+01 | Cr | 6.05E+05 |
| 237Np | 1.33E+02 | 151Sm | 3.35E+06 | U Total | 6.02E+05 |
| 241Pu | 1.25E+05 | 93Zr | 4.42E+03 | Zr | 4.09E+05 |
| 240Pu | 1.22E+04 | 243Cm | 1.24E+01 | Bi | 5.61E+05 |
| 99Tc | 2.85E+04 | 79Se | 1.32E+02 | Ca | 2.55E+05 |
| 232U | 4.25E+01 | 126Sn | 6.00E+02 | Cl | 8.64E+05 |
| 125Sb | 2.49E+04 | 236U | 5.92E+00 | TIC as | |
| 231Pa | 2.72E+02 | 113mCd | 1.65E+04 | CO3 | 9.80E+06 |
| 59Ni | 1.37E+03 | 93mNb | 3.86E+03 | TOC | 1.27E+06 |
| 155Eu | 7.69E+04 | 232Th | 8.12E+00 | PO4 | 5.32E+06 |
| 241Am | 1.43E+05 | 238U | 2.01E+02 | NO3 | 5.48E+07 |
| | | **TOTAL** | **1.94E+08** | NO2 | 1.21E+07 |
| | | | | SO4 | 3.66E+06 |
| | | | | **TOTAL** | **1.51E+08** |

Source: Tank Waste Inventory Network System, Best Basis Inventory, September 2003.

contaminate water supplies for thousands of years. Waste tanks also contain more than 1,000 kilograms of plutonium-239.[19]

Hanford's waste tanks contain complex mixtures that fit into 89 separate chemical profiles.[20] Chemical concentrations in each of the tanks widely vary by as much as 100 percent.[21] Sodium (Na+) makes up approximately 80 percent of the cationic content by weight, followed by aluminum (Al) at around 5 percent wt. There are also large concentrations of cations from construction materials such as iron (Fe3+), nickel (Ni2+) and chromium (Cr3+).[22] The dominant chemical anion in tanks is nitrate (NO3), which constitutes about two-thirds of the weight. Other abundant anions include hydroxides (OH), nitrites (NO2), and carbonates (CO3-2), phosphate (PO4-3), chlorine (Cl), fluoride (F), silicates (SiO4-2), and sulfates (S04-2).[23]

Although radioactive materials make up about one percent of Hanford's waste volume, they are enough to make the wastes highly dangerous, with exposure levels inside the tanks as high as 10,000 rad per hour.[24] There are

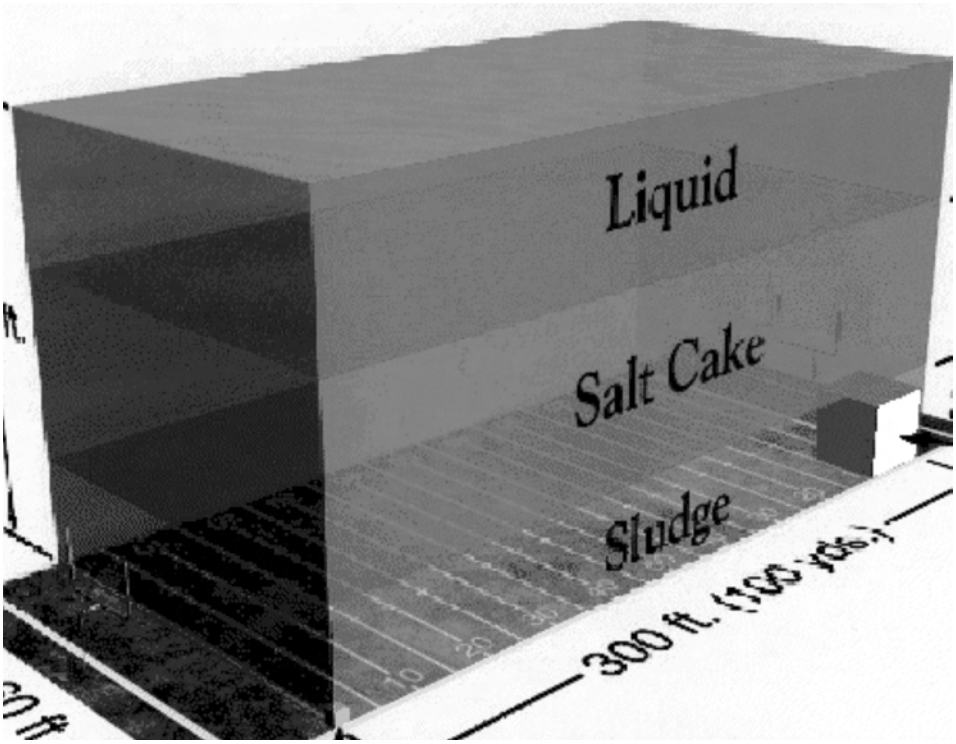


**Figure 1:** Volume and Radionuclide concentration in Hanford HLW. Soluble wastes are ~80 percent of the volume and contain ~50 percent of the total radioactivity. About 96 percent of the radioactivity in soluble wastes is cesium-137. Insoluble sludge contains ~95 percent of the total Sr-90, and >90 percent of the total transuranics. (Adapted from: DOE/RL-98-34.)

several forms and layers of wastes, which are "heterogeneous in all phases, both within a given tank and among different tanks."[25] (See Figure 1). Generally, the wastes are in three basic forms.

- **Sludge:** A dense, water insoluble component that has settled to the bottom of the tank to form a thick layer of varying consistencies;
- **Saltcake:** A crystallized salt waste formed above the sludge, which is mostly water soluble; and
- **Liquid:** Above or between the denser layers and sometimes embedded in saltcake are liquids of water, dissolved salts, and other chemicals called supernate.

The basis for high-level waste management at Hanford was established in World War II to meet production deadlines and limit waste storage costs. Because wastes coming out of the reprocessing plants were acidic, the U.S. government decided to neutralize them by adding sodium hydroxide (lye) and water so that cheaper carbon steel could be used to line the tanks, rather than more expensive high quality stainless steel. The decision to maintain a high PH, to reduce corrosion of the steel liners, substantially increased the volume of wastes.[26]

The wastes are stored in two general types of tanks (see Figures 2 and 3).

**Single Shell Tanks (SST)**—There are 149 SSTs with a single 1/4-inch carbon steel wall liner surrounded by concrete. They range in capacity from 55,000 to 1 million gallons, and were built between 1943 and 1964. The SSTs are

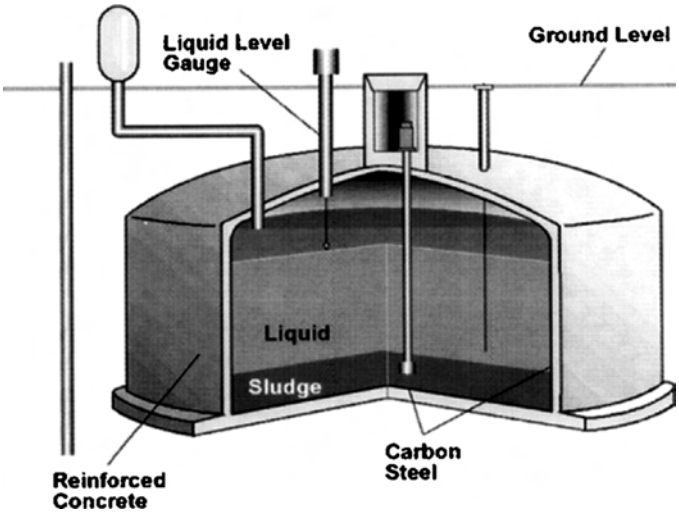


**Figure 2:** Single-Shell Tank. (Source: Pacific Northwest National Laboratory.)

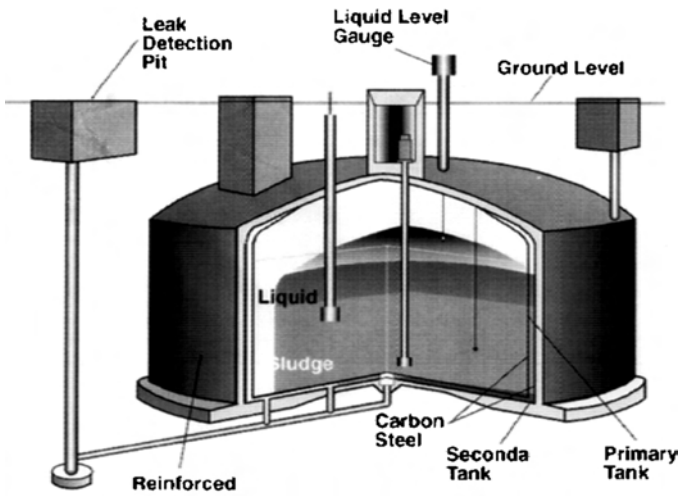


**Figure 3:** Double-Shell Tank. (Source: Pacific Northwest National Laboratory.)

clustered in 12 "Tank Farms." No wastes have been added to the SSTs since 1980. Of these, 67 tanks are estimated to have leaked over 1 million gallons.[27] The single-shell tanks store about 132,500 cubic meters of saltcake, sludge, and liquid containing 110 million curies of radioactivity.[28] About 90 percent of SST wastes are sodium nitrates and nitrites. About 75 percent of the radioactivity in these tanks comes from strontium-90 concentrated in sludge and 24 percent from cesium-137 in soluble liquids and saltcake.

**Double-Shell Tanks**—Between 1968 and 1986, 28 tanks with double steel liners, were constructed with a capacity of 1 to 1.16 million gallons. They contain about 83,279 cubic meters or 23 million gallons of mostly liquids (~80 percent), as well as sludges and salts.[29] The estimated amount of radioactivity in the DSTs is about 80 million curies (Cs-137 = 72 percent and Sr-90 = 27 percent).[30] As in single-shell tanks, wastes are primarily composed of sodium salts, and also have additional metal hydroxides, phosphates, carbonates and sulfates. None of the DSTs has leaked, but at least one has experienced major corrosion problems.[31] A technical basis for controlling corrosion of double–shell tanks through chemistry controls has thus far proven illusive.[32]

## CONSEQUENCES OF THE PRODUCTION IMPERATIVE

Efforts to keep waste storage expenses down during the period of nuclear weapons production, created significant problems. Approximately 120 to 130 million gallons of high-level wastes were discharged to the ground.[33] Wastes were transferred extensively, between tanks, without adequate

**Figure 4:** Diatomaceous earth in Hanford Tank 104-U (capacity, 530,000 gallons). (Source: Pacific Northwest National Laboratory.)

documentation,[34] and with little regard for chemical compatibility, heat loads or radioactive concentrations. Nearly 300 chemicals and chemical products were added during the course of waste processing,[35] including at least 5,000 tons of organics.[36] Additionally, hundreds of tons of cement,[37] and diatomaceous earth,[38] (see Figure 4) were dumped in several tanks. Wastes were evaporated, permitted to boil, and corrosion combined with the settling of high-heat sludges at the bottom of tanks resulted in the failure of steel liners.

These practices now pose major unresolved questions about Hanford's waste characterization, affecting safety and disposal. Subsequently, this situation was made worse by decades of neglect, causing more than a third of Hanford's tanks (67 SSTs) to leak high-level wastes, some that reached groundwater which eventually enters the Columbia River.[39]

In the summer of 1990 several Congressional and DOE investigations identified serious safety concerns regarding risks of explosions and fires in Hanford's high-level waste tanks.[40] All told, 60 tanks were placed on a "watch list" required by Congress, which were reviewed and analyzed. By August 2001, the Energy Department announced it had resolved all HLW tank safety issues[41] that surfaced earlier.[42]

Around the same time, however, the U.S. Nuclear Regulatory Commission warned that Hanford's HLW tanks "represent immediate concerns" particularly because of aging and deterioration.[43] The emphasis on safety of waste processing, NRC pointed out, should not overshadow the waste tanks because of "considerable environmental and public risk posed by continued operation

of the tanks with their associated leakage and potential for collapse and explosion."[44]

All of Hanford's HLW waste tanks generate potentially flammable gases.[45] Radiation, decay heat and chemical changes in the wastes generate toxic, flammable, and potentially explosive gases, such as hydrogen, nitrous oxide, ammonia and volatile organics, which can build up in the wastes and be rapidly released during retrieval.[46] NRC-sponsored research indicates that "even very small releases can collect in equipment or in poorly ventilated tanks and result in a flammable gas hazard."[47]

One of Hanford's most troublesome and, perhaps, most dangerous tanks, SY-101, continues to present potentially significant risks, decades after its dangers were first discovered. In October 2003, after the tank was declared safe and wastes were added several months earlier, SY 101 was reported to have "the propensity to undergo a large buoyant displacement gas release event and has sufficient retained gas [hydrogen] to achieve 100 percent of the lower flammability limit."[48]

## NO ROOM AT THE REPOSITORY?

By 1990, the DOE announced its basic goal was to process and dispose of high-level wastes (HLW) in all of Hanford's 177 tanks. However, it soon became apparent that geological disposal of all of Hanford's high-level wastes would result in the production of some 220,000 glass logs,[49] which increased waste shipments, and potential costs. The 1982 Nuclear Waste Policy Act imposes a limit of 70,000 MTHM limit on the proposed Yucca Mountain site.[50] If that amount is exceeded, the law requires a second repository to be selected. DOE spent fuel and high-level wastes are to make up no more than 10 percent of this limit.

Reducing the geological disposal of high-level wastes involves a complex system of waste fractionation[51] and multiple ion-exchange processes,[52] which were incorporated into Hanford's Tank Waste Remediation System (TWRS) in 1996. First, soluble liquids, and salts, comprising more than 80 percent of the total volume—which DOE calls "low-activity" (LAW) wastes—are to be separated from the remaining "high-level" waste sludge. Soluble wastes contain about half of the total radioactive inventory including about 96 percent of the total cesium-137 and the bulk of several long-lived radionuclides such as technetium 99, selenium 79, iodine 129, and carbon 14. Insoluble tank sludge contains about 95 percent of the total strontium-90 inventory and more than 90 percent of the long-lived transuranics.

Using separations technologies, DOE was to remove at least 98 percent of the radioactivity from soluble wastes to allow for their disposal onsite.[53] The treated insoluble sludges were to be combined with the separated radionuclides

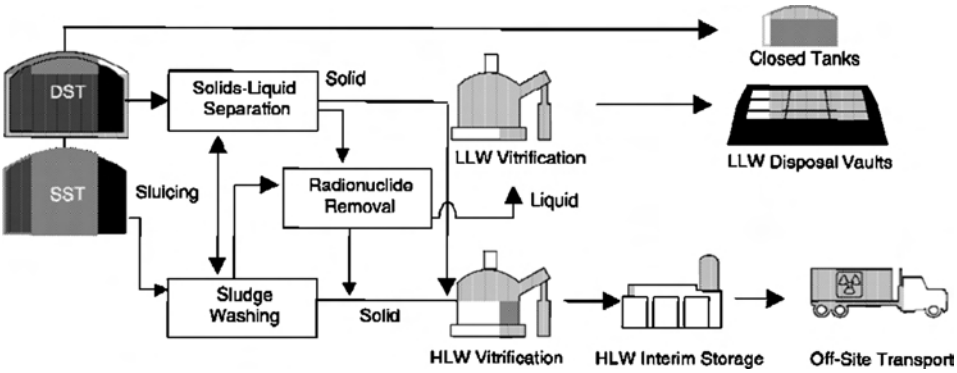


**Figure 5:** Simplified flow sheet for the Tank Waste Remediation System (TWRS) 1996–2002. (Source: NAS Research Needs for High Level Wastes Stored in Bins and Tanks at U.S. Department of Energy Sites, 2001.)

from LAW processing and vitrified in the HLW glass melter and would be stored on site to await geological disposal. Decontaminated "low-activity" waste would also be rendered into glass.[54] As a result, the TWRS project was expected to generate approximately 14,500 high-level glass canisters (15,700 cubic meters) and more than 100,000 low-activity glass packages (271,000 cubic meters).[55] (See Figure 5.)

In February 2004, however, DOE stated that "Yucca Mountain does not have the space for all defense HLW waste."[56] In order to accommodate the burgeoning inventory of spent reactor fuel,[57] DOE has decided to reduce the amount to be disposed to less than half of the glass logs expected to be generated for all DOE high-level wastes.[58] Assuming a proportional cut in disposal, Hanford's allocation will be reduced by over 60 percent. Thousands of high-level waste canisters are expected to remain at Hanford and other sites, awaiting disposal in a second repository. (See Figure 6.)

DOE's decision to curtail geological disposal of defense HLW is derived from hypothetical assumptions made in 1985 that a typical canister produced at the

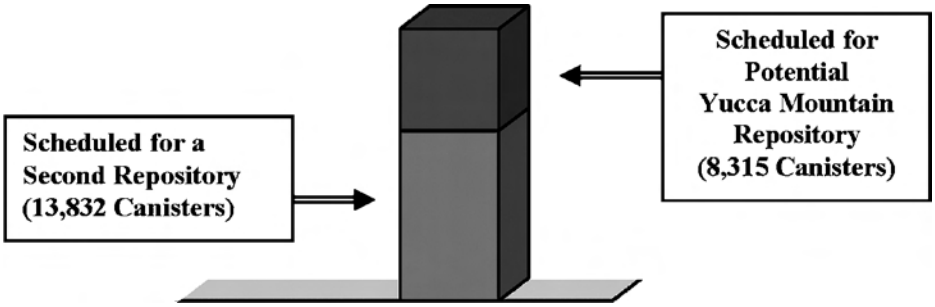


**Figure 6:** Projected disposal of DOE High-Level Waste canisters scheduled for a second repository (13,832 Canisters), scheduled for potential Yucca Mountain Repository (8,315 Canisters). (Source: DOE/EIS-0250, Appendix A.)

Savannah River Site would be the equivalent of 0.5 MTHM.[59] Since defense high-level wastes have nearly all uranium removed as a result of reprocessing, it is difficult to make comparisons based on the uranium content in commercial spent reactor fuel. Given this problem, the DOE assumed that each canister would contain 150,000 Ci.[60] Based on this formula, DOE estimated in 1985, that approximately 21,000 canisters would be "approximately equivalent to 10,000 MTHM of commercial HLW."[61]

DOE's assumption of the total number of canisters to be sent for disposal in a repository was also "based on in situ disposal of older wastes which are not readily retrievable from the 149 single-shell tanks."[62]

Actual production data show that DOE's criteria are not supported by current waste loading in HLW glass canisters at SRS. Since 1996, some 1,500 canisters produced at the Savannah River Site each contain an average of 3,500[63] curies to 10,500 curies.[64]

Risk-based criteria, based on radioactive concentration or radiotoxicity were identified by the National Research Council in 1999, which would allow disposal of "the complete inventory of DOE HLW."[65] DOE concurred in 2002, finding that disposal of all projected HLW canisters "would not change the cumulative impacts."

Under the Final Environmental Impact Statement for the Yucca Mountain, issued in 2002, the repository does not have space limitations that would prevent the disposal of 22,100 canisters, but DOE has chosen to ignore risk-based approaches to defense HLW allocation because they "would change the number of canisters . . . analyzed for the Proposed Action."[66] Based on the current average radionuclide concentration in HLW canisters produced at the Savannah River Site, the total number of canisters and shipments to Yucca Mountain could be substantially larger, with commensurate cost increases.

In 1996, National Research Council noted that technical factors, would not limit defense high-level waste disposal in Yucca. "Since the repository capacity is specified in tons of heavy metal equivalent, [disposal of 220,000 canisters] may not seriously affect the rules for eventual disposal in a geological repository." However, "their large number would surely exacerbate problems . . . which in turn, would present challenges to public acceptability."[67]

In its Record of Decision, DOE fails to address major inconsistencies in the 1985 criteria used to justify limited disposal of defense high-level wastes. While DOE concedes that all projected HLW defense canisters can be disposed in the potential Yucca Mountain disposal site, using criteria based on radionuclide concentration and toxicity, DOE has not provided quantifiable arguments against using these criteria. It may be that operational and disposal costs are high; or that there are physical and social obstacles that limit defense HLW disposal in the potential Yucca Mountain site. However, these concerns are not articulated in DOE's policy documents limiting disposal of defense high-level

radioactive wastes. Rather, DOE appears to rely on outdated assumptions and vague assertions.

## MAJOR CHALLENGES

### "Fast-Tracking Safety"

Vitrification of Hanford's high-level wastes requires a high degree of safety, particularly since it is the largest, first-of-a-kind project and also the largest project of its kind in the world.[68] (See Appendix A.) The waste treatment plant involves processing of tens of megacuries of radiochemicals, posing potential risks of leaks, nuclear criticalities, explosions, fires and large environmental releases.[69] The U.S. Nuclear Regulatory Commission considers the Hanford high-level waste vitrification plant as having radiological inventories and accident consequences comparable to a nuclear power plant.[70] Key radionuclides considered as exposure hazards during processing include carbon-14, strontium-90, iodine-129 and cesium-137.[71]

Until 2001, the NRC was in the process of establishing safety regulation of the Hanford high-level waste vitrification plant through a Memorandum of Understanding, signed in January 1997.[72] Because the vitrification plant was to be privately owned, the MOU was intended to develop a regulatory program that would allow for the transition to NRC regulation.

In assuming this new responsibility the NRC encountered major differences in safety regulation between NRC and DOE. For instance, DOE self-regulates safety primarily through a system of "Orders,"[73] which are not on their own, legally binding, but rather are enforced as contract requirements. Under DOE cost-plus contracts, the DOE must pay for any additional costs for compliance with safety orders. Since they are not subject to the Administrative Procedures Act, DOE Orders can be changed at individual sites, without public knowledge or involvement.

The Defense Nuclear Facilities Safety Board (DNFSB) was created in 1988 and provides independent oversight of DOE defense nuclear facilities.[74]Since its inception the DNFSB has been involved in safety issues pertaining to Hanford's high-level waste activities. The DNFSB does not have regulatory authority and can only make recommendations to the Secretary.

By contrast, NRC has a well-developed system of formal regulations that have the force of law, are subject to the Administrative Procedures Act, and are issued to licensees as mandatory requirements.

However, in May 2000, DOE ended privatization at Hanford for cost reasons.[75] The DOE subsequently terminated NRC's involvement and reestablished self-regulation under its traditional, cost-plus, management and operations (M&O) contract system.

In its June 2001 report the NRC identified over two dozen significant safety issues and over 50 specific topics in the current design and approach which remained to be resolved.[76] "Several scenarios involving large radiochemical inventories (in tanks), flammable gases, organic ion exchange resin interactions, glass melters, and cold chemical effects," according to the NRC, "were found to have potential accident consequences to the workers and the public of significant severity and risk."[77]

The NRC found that plant "has more stored chemical energy for prompt potential events directly involving the radionuclides in their mobile forms,"[78] and thus, radiological consequences to members of the public could result in doses in the hundreds or thousands of rem.

In arriving at this conclusion, the NRC was actively involved in the development of the Documented Safety Analysis (DSA), a required safety document that extends to the design, construction, and operation of a nuclear facility.[79] The DSA includes a comprehensive hazard analysis associated with accident scenarios that could result in significant consequences to members of the public and the environment. In turn, the hazard analysis is required by regulation to include estimates of the frequency of unmitigated risks, which do not take into account preventative features that would lessen the consequences of an accident, "other than initial conditions and the basic physical realities of a given operation."[80] This approach is supposed to envelope uncertainties that provide adequate safety margins.

The NRC found that DOE did not "appear to adequately address the significance of unmitigated events." DOE's "implicit assumptions" would result in "less severe consequences" and "may result in overlooking and not identifying safety controls and their requirements, including reliabilities." NRC reported "there still is an apparent bias . . . to implicitly rate hazards in a mitigated manner . . . Thus, it is not clear that safety requirements are being adequately identified and categorized."[81]

In this context, the NRC estimated the total unmitigated risk of major accidents involving large radiation releases, such as a melter steam explosion or a resin fire, at the Hanford vitrification plant was 2.4E-2/yr (annual risk of 2.4 percent).[82] This translates into a 50–50 chance of a major accident over 28 years of operation.[83] (See Table 2.)

Chemicals also pose significant hazards. Tank failures containing nitric acid and anhydrous ammonia could cause severe injuries and death and "render the facility uninhabitable" to an area extending beyond a mile.[84]

However, NRC found that (with the exception of the glass melters) existing "mitigation methods exist that are compatible with the regulations and offer the potential for reducing process accident risk to more acceptable levels (circa 2E-6/yr)."[85]

A melter steam explosion constituted more than 50 percent of the unmitigated risk of a catastrophic accident[86] and NRC staff expressed concerns

**Table 2:** Unmitigated accidents at the hanford high-level waste treatment plant.

| Event | Unmitigated consequence impact, receptor at 100 meters, rem | Part 70 consequence category | Estimated frequency (uncontrolled event/yr) | Likelihood (probability) bin | Unmitigated impact risk, yr-1 |
|---|---|---|---|---|---|
| LAW tank failure | 3,000–6,300 | High | 2E-5 | Unlikely | 3E-5 to 6E-5 |
| HLW tank failure | 6,000–12,000 | High | 2E-5 | Unlikely | 6E-5 to 1.2E-4 |
| Cesium tank—loss of cooling/boiling (1,000 gal) | 25,000 | High | 1E-6 | Unlikely | 1.25E-3 (30 percent of total unmitigated risk) |
| Melter/canister failure, cold cap dispersal | 14,500 | High | 1E-3 | Unlikely | 7E-4 (3 percent of total unmitigated risk) |
| Cesium eluting, resin/nitrate interaction | 3,400 | High | 1E-3 | Unlikely | 1.7E-3 (7 percent of total unmitigated risk) |
| Hydrogen deflagration/ LAW heel | 20,000 | High | 1E-5 | Unlikely | 1E-4 |
| Melter/Steam explosion | 26,000 | High | 1E-3 | Unlikely | 1.3E-2 (54 percent of total unmitigated risk) |

Source: NUREG 1747, Table 4.

that "few tests appear to be planned to verify safety parameters prior to construction."[87,88] NRC's concluded that DOE and its contractors had sufficient knowledge and capabilities to mitigate the likelihood probability to 1E-4.[89]

"Further analysis" was required, however, to determine if melter risks could meet probabilities acceptable under NRC regulations for reactors and fuel cycle facilities (10 CFR 70).[90]

Based on review of the nine high-level radioactive waste and several low-level and mixed waste vitrification facilities throughout the world, NRC-sponsored research points out that "operating limits on chemical composition, redox control, and glass properties such as viscosity, electrical resistivity, phase separation and liquidious temperature should be established before start of the radioactive process."[91] Failure to meet these conditions have led to serious problems. For instance, since 1991, there have been eight melter-related incidents and failures in the DOE complex.[92,93] (See Table 3.)

On one instance, on 21 April 1996 pressurized steam vented rapidly through the melted glass the Oak Ridge low-level, In Situ Vitrification (ISV) plant and caused an explosion that expelled 20,000 kgs of glass, spewing hot fragments over 100 meters from the melter site.[94]

Of major concern to the NRC was that, proposed designs "do not consider prevention and controls [and] do not include important auxiliary effects in the analyses, such as common mode failures, operability, recoverability, and plant habitability for operators"[95] The NRC concluded that "regulatory and safety issues associated with a much larger facility do not appear to have been considered . . . On many occasions, there was an implication that regulatory reviews were not allowed to impact cost and schedule . . .[96]

However, since the NRC ended its relationship, DOE has taken steps to "reduce conservatism" in its high-level waste safety controls at Hanford to "allow work to be performed more quickly."[97] As a result DOE and its contractors have significantly curtailed safety analyses and oversight, reduced operational safety procedures, and eliminated DOE approval of important changes in safety analyses and subsequent construction decisions.

Efforts to "reduce conservatism" have now, however, proven to be costly and time consuming. In 2003 the Defense Nuclear Facility Safety Board took issue with the design assumptions about earthquakes stating that the Hanford site could experience destructive seismic activity 15% greater than California sites.[98] In March 2005, after subsequent testing, the Energy department was compelled to suspend construction for facilities that would handle a preponderance of wastes and to increase the design standard from 20 percent to 40 percent.[99]

DOE's preference for administrative over engineering controls, because they cost less is also of concern. According to NRC, DOEs approach "appears to rely extensively on operator actions to prevent or mitigate the

**Table 3:** Summary of melter-related incidents in the DOE.

| Incident | Corrective action | Lessons learned |
|---|---|---|
| *Savannah River DWPF* Wicking of the glass stream during pouring resulted in plugging of the discharge orifice on a regular basis. (1997) | The discharge shut was modified to reduce wicking and in addition, remote equipment was installed to clean a plugged orifice. | Future melter designs should account for wicking of glass pour streams. |
| *West Valley WVDP* The transposition of a weld symbol at the dam and trough interface in engineering drawings was the root cause for glass seepage onto discharge wall. Missing weld resulted in separation between dam and trough. (1996) | Since the incident occurred during cold operations, hands on repairs were conducted, and operations were subsequently resumed. | Rigorous design review check and control should be implemented. |
| *West Valley WVDP* The ceramic nozzle liners failed due to insufficient thermal expansion allowance. (1997) | The nozzle liners were redesigned. | Selection of materials and design of components should undergo evaluation prior to radioactive operations. |
| *West Valley WVDP* Formation of glass fibers in the discharge section led to the blockage of the discharge orifice. This was the result of high air-inflow through the discharge orifice to the melter.(1996) | Flow-reducing orifice was installed to reduce airflow. | Operating limits for airflow rates, pressure, and temperature should be established prior to start of process. |
| *Savannah River* The mixed low-level radioactive waste vitrification facility in the M-area suffered an electrode failure that caused accelerated corrosion and failure of molybdenum electrodes. This was partially attributed to the failure of cooling systems for the electrodes. (1997) | Melter replaced. | Corrosivity of the melt and its compatibility with the components should be established before melter operations. Performance of the melter should be continually assessed during operations via quality assurance programs and safety audits. |
| *Fernalds* A nonradioactive melter failed, dumping 6,000 kg of glass on the floor due to degradation of the melter components caused by incompatible feed chemicals. (1996) | The facility was shut down. | Corrosivity of the melt and its compatibility with the components should be established before melter operations. Performance of the melter should be continually assessed during operations via quality assurance programs and safety audits. |

*(Continued on next page)*

**Table 3:** Summary of melter-related incidents in the DOE. *(Continued)*

| Incident | Corrective action | Lessons learned |
|---|---|---|
| *Oak Ridge National Laboratory* A radioactive waste In situ Vitrification Plant experienced a steam explosion which resulted in the release of off gas and an expulsion of 20,000 kgs of molten glass, spewing fragments 100 m from the melter site. (1996) | Recommended corrective actions included diversion of standing water around the pit, installation of flow-monitors and curved vent pipes beneath the melt to provide alternate paths for steam, submelt pressure measurement, and video monitoring of the melt surface. | Melters should have safeguards designed to account for not only normal operating conditions, but also for abnormal conditions such as steam explosion. |
| *Hanford* A large-scale test of In situ Vitrification on a buried 6000-gal tank resulted in a steam explosion which raised the off-gas hood 12 in. from the ground and the expulsion of molten soil. (1991) | The facility was shut down. | The cause was ascribed to sealing of the walls of the tank to the melt body precluding normal pathway for dissipation of steam from the melt. |

Sources: Vijay Jain, Process Safety Issues Associated with Melter Operations During Vitrification of Radioactive Wastes, Proceedings of the XVIII International Congress on Glass, 2000, and ORNL/ER-371.

effects of chemical hazards. . . . The normally accepted practice and NRC regulatory emphasis are minimization of the reliance upon administrative controls."[100]

A key safety concern where engineering controls are important is fireproofing and fire suppression. The Waste Treatment Plant will be handling large quantities of flammable materials. However, DOE and its contractors are cutting costs by reducing steel fireproofing and fire suppression requirements. In 2000, the NRC objected to this approach because it would "severely limit any future modifications" and "lack of fire suppression capability along with lack of steel protection in the same area makes administrative control of combustibles the only defense measure . . ."[101]

In April 2003, the Office of Environmental Management's Director of for Safety and Engineering at DOE's Headquarters, reiterated the NRC's concerns regarding a decision made by ORP to reduce fire protection requirements at the Waste Treatment Plant.[102] It was pointed out that one of the areas, which was given a "low" combustible rating would have to withstand fires that are "equivalent of 2,370 pounds of wood in an area the size of an individual office." Moreover, "the construction contractor has also proposed a combustible decontamination coating in lieu of stainless steel on all surfaces which will appreciably add to the combustible loading in these spaces."[103]

Like the NRC, the headquarters review found that "administrative controls are not an approach that would be intentionally selected for new facilities

during construction where complying with the standards is relatively easy and could avoid controls."[104]

In January 2004, however the Office of River Protection (ORP) concurred with the contractor to reduce fireproofing in its design approach. According to the staff of the Defense Nuclear Safety Board, "it is not known how the analysis will address fires when the sprinkler suppression system is inoperable."[105]

Finally, the growing number of worker exposures and injuries in the Hanford tank farms, and construction mistakes over the past two years provide warning of potentially more serious problems to come. Over the past two years, despite admonitions from DOE researchers about occupational dangers,[106, 107] several workers have been exposed to tank vapors. For instance, in July 2003, 12 workers breathed in radioactive materials, and contaminated their skin, while working in a pit near a high-level waste tank. "The health physics technician counting contamination samples unsuccessfully tried to stop the work,"[108] despite the high number of workers being put at risk. Some 90 workers have reported illnesses and injuries to site medical professionals, claiming they were caused by from exposure to tank vapor exposure.[109] This has resulted in investigations by the State of Washington, the U.S. Congress and the DOE's Office of Inspector General.[110, 111]

In February 2005, Waste Treatment Plant construction workers reported to DOE "a chilling effect with regard to fear of retaliation for reporting safety, medical, and labor relations issues. Approximately 20% of the workers interviewed described harassment, intimidation, and fear of termination when using the first aid facility on the site or after using a private physician and an equal percentage voiced the belief that when individuals raise safety concerns, those individuals are targeted for future lay off lists."[112]

## ONSITE DISPOSAL

DOE's accelerated cleanup program will result in direct on-site disposal of a substantially larger amount of radioactivity from Hanford's high-level waste tanks than agreed by the NRC staff in 1997. In an effort to reduce the amount of wastes to be processed in the vitrification plant, contents from dozens of Hanford tanks are to be processed, without radionuclide separation, using bulk vitrification and possibly other "supplemental" technologies[113] for permanent on-site disposal, leaving behind substantially larger amounts of radioactivity on-site.

To accomplish its objectives to reduce the number of HLW canisters and to leave more radioactivity on-site, DOE is seeking to reclassify high-level wastes as "incidental."[114] In July 2003, a federal district court ruled that DOE does not have the authority to reclassify high-level wastes.[115] The following year Congress enacted legislation which authorized the DOE to self-regulate HLW disposal with NRC consultation. However, Hanford was excluded from this

provision. After vigorous protest by Washington State's US Senate Delegation. Also, sludge from deteriorated spent reactor fuel at the Hanford K-basins[116] and wastes from a dozen tanks[117] are designated as "potential transuranic wastes" for disposal in the DOE's Waste Isolation Pilot Project (WIPP) in New Mexico. Attempts to dispose of Hanford HLW tank wastes implicitly raises the question: Will WIPP, by default, become the second repository for thousands of HLW canisters which DOE claims it has no room for in Yucca Mountain?

Concurrent with the design and construction of treatment facilities, 40 tanks are scheduled to be emptied and "interim" closed within the next two years.[118] Such tanks are expected to have all retrievable wastes removed and to be in a stable state for final closure.[119] Once wastes are removed, cement will be poured in the tanks to immobilize yet-to-be determined concentrations of residual long-lived radionuclides.

The scientific underpinning for disposal decisions at Hanford should be a sound understanding of the fate and transport of tank wastes in the environment. Even though a large amount of wastes were discharged or leaked into the soil, DOE's current understanding of contaminant mobility "is inadequate to fully support cleanup, closure, or performance assessment-related decisions."[120]

The closure of 177 large tanks and many miles of underground pipes and related infrastructure will leave behind significant amounts of residual high-level wastes. According to DOE-sponsored research, radionuclides from tank closure represent "one of the most significant long-term dose contributors on site . . . However, the radionuclide release rate from these solids is virtually unknown."[121]

Under current regulations, the NRC still determines what constitutes high-level wastes for geological disposal at Hanford.[122,123,124] However, NRC has chosen to exercise its authority through staff-level agreements with the DOE. NRC has yet to issue a formal determination by rulemaking, or other means, regarding on-site disposal of defense high-level wastes. This regulatory approach has allowed DOE to proceed with actions, such as disposal of HLW tank residuals at the Savannah River Site, which for all practical purposes are irreversible. It also, in the case of Hanford, allows DOE to disregard agreements with NRC staff, without regulatory consequences.

Given these circumstances, the NRC staff provisionally agreed to a plan by DOE in 1997, to remove radionuclides from soluble high-level wastes to allow their on-site disposal.[125] This agreement was specifically based on estimates provided by DOE that:

1. Radionuclide removal to the maximum extent technically and economically practical will leave no more than 9.8 MCi Cs-137(including barium-137 m decay product)and 6.8 MCi Sr-90 (including yttrium- 90 decay product) low activity wastes.[126]

2. Removal of TRU as required . . . will ensure all solidified LAW is <100 nCi TRU/g.

3. All disposal requirements including those defined by the performance assessment required by DOE Order will be met.[127]

NRC staff found that DOE's plan "is not sufficient to make an absolute determination at this time."[128] Moreover, if DOE did not utilize separation technologies embodied in the Tank Waste Remediation System,[129] and if there were large increases in tank inventory data, "the incidental waste classification must be revisited by DOE and the NRC consulted."[130]

Based on activities outlined in the Integrated Mission Acceleration Plan for the processing and disposal of Hanford's high-level wastes,[131] the Energy department is:

- seeking to dispose of substantially larger quantities of radionuclides than agreed to by the NRC staff; (See Table 4)
- proceeding to dispose of wastes with significantly greater radionuclide inventories than provided to NRC staff; and
- failing to demonstrate compliance with waste performance assessments.

DOE has identified wastes in 62 tanks, which are to be retrieved and immobilized using "supplemental" technologies without additional removal of radionuclides.[132] Waste from these tanks combined with decontaminated low-activity wastes coming from the treatment plant could result in the onsite disposal of more than three times the strontium-90 and over than six times the transuranics agreed to by the NRC staff.

Waste inventory data, particularly for transuranics have increased since DOE entered into the 1997 agreement with the NRC. As Figure 8 indicates transuranics increased nearly three-fold. (See Figure 8)

Relative to meeting the terms of the 1997 agreement to perform waste performance assessment, a major concern is the groundwater impact from iodine-129. DOE was informed by CH2MHILL, Hanford's HLW tank farm contractor, in September 2003, that: "iodine is a key driver in the risk assessment and the inventory of iodine is uncertain for tank waste and secondary waste."[133] Iodine-129 is of concern, because of its very long half-life and potential to harm the human thyroid. DOE estimates that five curies of I-129 from the disposal of secondary processing wastes are a dominant dose contributor.[134] In January 2004, the DOE provided it first performance assessment for on-site low-activity waste disposal, in which it indicates that on-site disposal of Hanford tank wastes meets expected requirements.[135]

However, this amount would result in drinking water concentrations of iodine-129 that are 22 times higher than the EPA's maximum concentration limit (MCL), if proper methods are used.[136] Instead, DOE bases its estimates

**Table 4:** On-site radionuclide disposal in low-activity wastes at Hanford (curies).

| Radionuclide | On-site IAW disposal agreed by NRC staff in 1997 (a) | 62 tanks scheduled for supplemental waste treatment b) (c) | 115 tanks scheduled for waste treatment plant (b) | Accelerated cleanup 177 tanks total on-site LAW |
|---|---|---|---|---|
| Cesium 137 | 9,750,000 (d) | 10,900,000 (d) | 2,370,000 (d,e) | 13,300,000 (d) |
| Strontium-90 | 6,800,000 (d) | 15,500,000 (d) | 6,380,000 (d,f) | 21,900,000 (d) |
| TRU (g) | 10,000 | 46,000 | 18,840 (h) | 64,840 |
| Technetium-99 | <30,000 | <7,200 | <22,800 | <30,000 |
| Carbon-14 | <5,300 | <1,300 | <4,000 | <5,300 |
| Iodine-129 | <51 | <14 | <34 | <48 |
| Tritium | <10,000 | <4,100 | <5,900 | <10,000 |
| Tin-126 | <1,600 | <140 | <460 | <600 |
| Selenium-79 | <1,000 | <20 | <114 | <134 |
| Uranium | <1,000 | <150 | <850 | <1,000 |
| **Total** | **16,600,000** | **26,500,000** | **8,800,000** | **35,300,000** |

(a) WHC-SD-WM-TI-699 Rev. 2 (1996), P. 4-1. This estimate includes the disposition of wastes in all 177 double and single-shell Hanford tanks.
(b) Estimates derived from Tank Waste Inventory Network System, September 2003.
(c) Accelerated Retrieval and Interim Closure Schedule Table 4.3, Potentially Low Curie Low-Activity Waste Tanks Table 4.6 (This estimate excludes wastes in tanks C-104,106,107,S-105,106, and 112 scheduled to go to the Waste Treatment Plant, and is based on disposition of wastes in 62 SSTs.)
(d) Daughter products of Cs-137 (mBa-137) and Sr-90 (Y-90) included.
(e) WHC-SD-WM-TI-699 Rev. 2 (1996) methodology. 3 percent of Cs,Ba-137 inventory.
(f) WHC-SD-WM-TI-699 Rev. 2 (1996) methodology. 3.78 MCi soluble Sr,Y-90 plus 3 percent of insoluble Sr,Y-90 inventory.
(g) Transuranic wastes as defined by the NRC.
(h) WHC-SD-WM-TI-699 Rev. 2 (1996) methodology. 9,600 Ci soluble TRU plus 3 percent of insoluble TRU inventory.

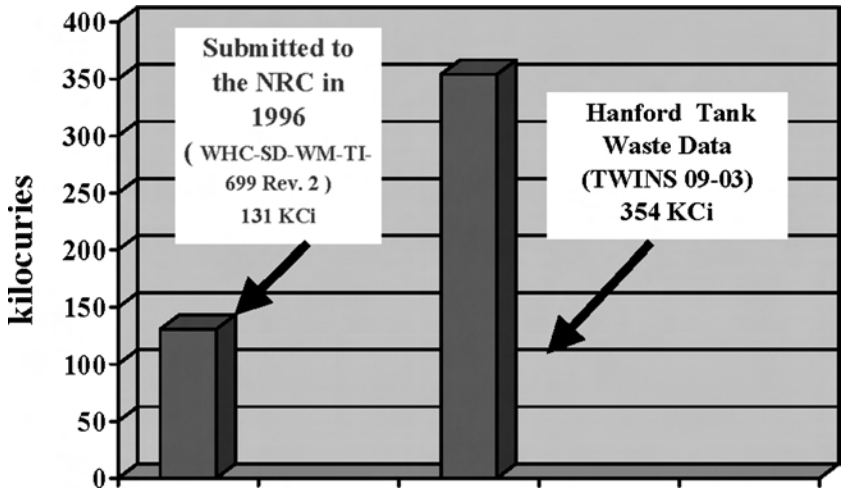


**Figure 7:** Increase of Transuranics in Hanford Tanks submitted to the NRC in 1996 (HC-SD-WM-TI-699 Rev. 2) 131 Kci Hanford Tank Waste Data (TWINS 09-03) 354 Kci Kilocuries.

on a whole body dose (MCL of 1pCi/L = 0.18 mrem/yr).[137] In doing so, DOE ignored a universally accepted principle that retained radioiodine concentrates almost exclusively in the thyroid, not the whole body. Moreover, DOE's dose estimation methodology ignores EPA standards and DOE's own orders.

## REDUCING SAFETY RISKS

The risks of high-level waste processing at Hanford, based on a preliminary estimate by the NRC, are comparable to those of U.S. manned space program.[138] As the largest, first-of-a-kind process handling large volumes of ultrahazardous materials, safety risks are compounded by inadequate waste characterization data and a lack of processing experience with actual wastes. Given these knowledge gaps, risk assumptions, design and construction decisions require strong elements of conservatism to envelope major uncertainties. Conservative parameters may overestimate risks, but result in a high margin of safety and public confidence.

Given the large risks involved, Congress should authorize the Nuclear Regulatory Commission to license the construction and operation of the Hanford vitrification plant; and certify the safety of stored high-level radioactive wastes. The NRC has invested three and half years at Hanford, and could effectively resume a transition to external regulation that can provide a consistent and comprehensive approach to tank waste storage and processing.[139] Legislation enabling NRC regulation should clearly define roles and responsibilities, such as:

- DOE retains title to all high-level radioactive wastes;

- NRC regulation of the vitrification plant would fall under 10 CFR Part 70 which governs nuclear fuel cycle facilities;
- The contractor would hold the license and DOE would not be subject to direct NRC fee assessment; and
- The certification of high-level waste storage tanks would be licensed under 10 CFR Part 76, because these are existing facilities being certified rather than licensed.

NRC has estimated that 32 full time staff equivalents would be required to carry out the work, which corresponds to $8 million per year. These costs are required to be recovered by charges levied on the licensee. Since NRC has already developed regulations and guidance, the annual costs may be lower for regulatory transition. Moreover, the total costs of NRC regulation are a small portion—less than half of one percent—of the total program cost.

## REDUCING ON-SITE DISPOSAL RISKS

DOE's efforts to dispose of substantially larger quantities of radionucides on-site from high-level wastes are premised on establishing "risk-based end states." Modeling of the natural attenuation of radionuclides over periods of hundreds to thousands of years is a limited approach that does not factor in:

- dramatic shrinkage of controlled areas at the Hanford site within eight years,
- accident scenarios involving the processing of high-level wastes,
- the high existing vulnerability of tribal people to environmental contaminants, and
- integration of natural resource risks with human health risks.

Over coming years, DOE plans to lift radiological controls over large swaths of the Hanford site for transfer to the U.S. Interior department's Fish and Wildlife Service. The transfer is intended to reduce DOE overhead expenses, while expanding the Hanford Reach National Monument.[140] More than 87 percent of the land DOE currently occupies will be shifted to Interior, by 2012 opening public access for thousands of people.[141] Even though potential vitrification accidents "show a significant distance effect,"[142] DOE's assumes that Hanford's current security perimeter of 6.8 to 9.3 miles (11,000 to 15,000 meters), will indefinitely serve as the boundary for public exposures.[143] To be more protective, NRC's guidance, which sets the public dose, including collocated workers, at the "fence line" of 100 meters should be adopted.[144]

Underscoring the need for public health conservatism is recent evidence indicating that tribal people living near Hanford are the most vulnerable to harm

from environmental contaminants. The Environmental Protection Agency reported in 2002 that fish in the Hanford Reach have the highest concentrations of contaminants in the Columbia River Basin, and that tribal people who eat fish from the Hanford Reach have up to a 1 in 50 lifetime risk of contracting fatal cancers.[145]

Because the tribal lifestyle is heavily dependant upon subsistence food gathering, protection of natural resources and human health are intrinsically linked. DOE has yet to make this connection. For instance, around the same time the EPA fish contaminant study was released, DOE set a standard limiting radiation exposure to fish in the Hanford Reach to no more than one rad (radiation absorbed dose) per day.[146] If a tribal adult eats fish so exposed from technetium-99, the annual human dose would be about 8.3 rems.[147]

As DOE seeks to transfer large parcels of the Hanford site to the Department of Interior, no comprehensive health and ecological risk assessments have been done as required under the Superfund Act.

Given these circumstances, NRC's 1997 provisional staff approval for the on-site disposal of 16.6 megacuries of radionuclides remaining in soluble tank wastes should be reconsidered, with the objective of significantly reducing this radiological contaminant burden. Towards this goal, the NRC should actively consult with the EPA, Washington State, Oregon and affected Indian tribes to establish comprehensive, formal limits on tank closure, HLW processing and disposal. A comprehensive health and environmental risk assessment of the Columbia River should be done in accordance with the Superfund Act.

Finally, the Nuclear Regulatory Commission should be funded by Congress to make a formal determination by rulemaking to allow on-site disposal of those high-level wastes which can be deemed as "incidental."

In DOE's haste to terminate its environmental mission, the Congress, federal and state regulators, and the Interior Department must actively ensure that DOE is not heeding advice that "sometimes the environmentally preferable course of action is to do little or nothing."[148]

## REDUCING PROJECT RISKS

DOE should cease its "fast-track" approach and follow numerous expert recommendations to build and operate "pilot" operations using actual Hanford high-level wastes. This was done at DOE's West Valley Vitrification Demonstration Project in New York, and would establish the necessary experiential basis for feed preparation, pretreatment and melter technologies. It was not done at the Savannah River Site for high-level, soluble waste pretreatment, which resulted in a 20-year failure costing $500 million with $1.8 billion estimated for a technological replacement. By virtue of the magnitude of the environmental, safety, and financial risks involved, processing of Hanford's high-level wastes is

of national importance and should have a commensurate level of project management attention by the Energy Department.

DOE has a history of failed projects, cost overruns, and delays,[149] which prompted the U.S. Congress in 1998 to seek the assistance of the National Research Council. The Council subsequently issued several reports[150,151] which found that:

- Environmental projects suffer from major delays and are about 50 percent more expensive than comparable federal and private-sector projects;
- Up-front project planning is inadequate;
- There is no consistent system for evaluating project risks; and
- DOE is not in control of many of its projects and had virtually abdicated its ownership role in overseeing and managing its contracts and contractors.

For over a decade, the DOE environmental cleanup program has been identified by the U.S. General Accounting Office as a "high-risk" program vulnerable to waste fraud and abuse. GAO describes DOE's management culture as one of "least interference" based on an "undocumented policy of blind faith in its contractors' performance."[152] The National Research Council also stresses a greater role by DOE:

> ...as the custodian of public funds, [DOE] should not abrogate to contractors project definition, acquisition strategy decisions, and project oversight. To effectively fulfill its project management responsibilities, DOE needs to expand its investment in human capital to develop a corps of qualified project managers commensurate with the value and complexity of its projects.[153]

As an environmental project unrivaled in its scope, risk, and expense, the management and oversight responsibility for the success of this project should not be abdicated to contractors. DOE must go well beyond its Cold War role of serving primarily as funding administrator.

DOE should begin by establishing a full-time multidisciplinary, Hanford HLW Project Management Group, reporting to the Assistant Secretary of Environmental Management. This approach is well established by the DOE's Office of Science and has been endorsed by the NAS as a proven way to enhance the success of large complex projects.

Concurrently, the pool of talent in the DOE and the private sector to carry out DOE's complex nuclear cleanup tasks is shrinking, which reduces competition and can negatively impact the successful outcome of multibillion-dollar high-risk projects.

In order to address these structural problems DOE should seriously consider establishing a special program to educate and train students in the necessary fields to prevent further erosion of key skills and knowledge that is fast

disappearing in the DOE complex. A model that DOE should consider is the one established by the Office of Naval Reactors, which recruits and pays for the education of qualified college students, in exchange for government service. The scope of this recommendation is well beyond the issues covered in this article but it, nonetheless deserves serious attention by the DOE and the Congress.

## APPENDIX A

### Technological Issues

The basic process to be deployed at Hanford of melting silica and adding materials to form glass has been around for some 2000 years. The melters to be used at Hanford are a joule-type with a ceramic lined furnace that is heated to a temperature of 1,150°C. by passing electric current through the glass by electrodes to produce borosilicate glass.[154] (See Figure 8.)

Worldwide, there are five high-level waste, and two low-level and mixed low-level waste vitrification plants that have used joule-heated melters.[155] While it is considered a mature technology with 20 years of experience, these types of melters have low glass production rates.[156] With an average design life of 3–5 years, several melters have experienced major problems, "such as breach of a

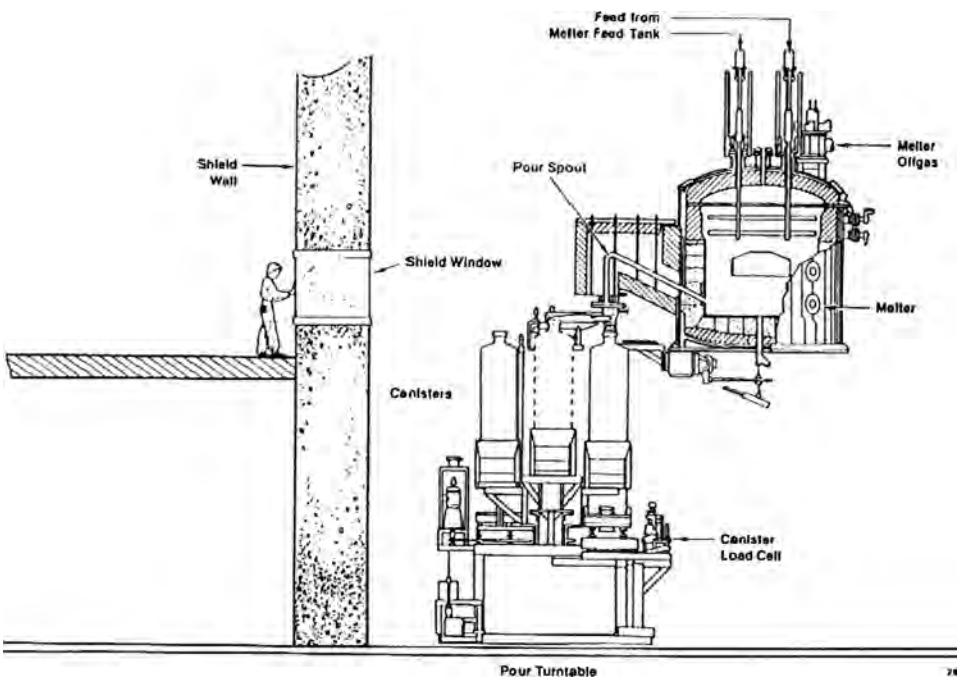


**Figure 8:** Handford waste vitrification plant melter/turntable. (Source: DOE/RL-900009.)

melter vessel, allowing molten glass to leak/drop out of the melter, uncontrolled off gas from the melter, failure of the joule-heating system, and plugging of the glass pour drain."[157]

The Hanford vitrification plant will have the largest melters in the world—two for high-level wastes and the other for low-activity wastes. Initially the TWRS program was to begin with a relatively small, pilot plant that would process between 6 to 13 percent of Hanford's tank waste. This would have allowed verification of design and technical approaches with minimal economic, programmatic, and safety risk.

For instance, according to NRC-sponsored research the use of surrogate wastes, as is the case with current Hanford melter tests, may prove inadequate because "actual Hanford wastes may be more reactive."[158] But the pilot plant was scrapped, despite recommendations by the NAS, NRC, GAO and DOE's construction contractor.[159] Instead the Energy department has decided to concurrently design and construct a full industrial-scale operation based on the concept of "learn by doing."[160]

As noted in 2001 by a DOE- sponsored review of nuclear waste vitrification melters: "Construction costs, although important, are not major determinants in life-cycle cost. . . . , High-level waste operating cost savings from increased waste loading or throughput may not be attainable without corresponding throughput improvements (or additional facilities) in retrieval, pretreatment, and low-level waste vitrification."[161] DOE's approach is already facing "potentially large cost and schedule overruns and performance shortfalls," predicted two years ago by the National Research Council.[162] For instance, in February 2004, it was discovered that an already installed waste processing tank did not meet safety inspection requirements after seven similar vessels were more than 94 percent fabricated, with similar flaws. Dozens of the welds in other waste processing tanks were also found to be "undersized or undercut, or have inadequate contouring." Apparently Bechtel and the fabricator did not check to see if tank construction comported with design drawings.[163] As a result of problems like these, estimated construction costs for the treatment plant have grown by more than 25 percent, from $4.35 Billion to $5.78 Billion.[164]

In the absence of pilot operations using actual Hanford Wastes, DOE faces several major challenges:

**Pretreatment.** This involves separation of radionuclides from soluble wastes, and chemical washing of insoluble tank sludges, prior to making feed for the melter. According to DOE, pretreatment "represents a significant portion of the HLW management costs and of the technical risk."[165] At Hanford, processes to remove corrosive metals, such as chromium, from tank sludge "are not effective."[166,167] The inability to remove sulfate adversely impacts low-activity waste glass production, which remains an important concern. Moreover, DOE has not been able to demonstrate that large-scale decontamination of soluble

wastes can work—as witnessed by the 20-year failure at the Savannah River pretreatment facility estimated to cost more than $2.3 billion in lost and future expenses.[168]

**Feed Preparation and Melters.** Preparing chemically balanced and homogeneous feed is of utmost importance because "the melter is quite unforgiving of batching errors."[169] This task is made difficult because knowledge of the characteristics of Hanford's wastes, according to the National Research Council, "is of little value in designing chemical remediation processing."[170] The inability to have proper feed can cause: (a) short-circuiting of melter electrodes by the phase separation of chromium, ruthenium, rhodium, and palladium from the melting glass;[171] (b) corrosion of the melter lining, clogging of the outflow of the glass melt to the canisters; (c) ruining the integrity of glass from chromium, phosphorus oxide, and sodium sulfate;[172] and (d) major accidental releases.[173] High radiation fields that require remote repairs and potentially frequent melter replacement,[174] exacerbate these problems.

**The Off-Gas System.** In effect, the melter serves to produce glass and as an incinerator which releases large amounts of contaminated carbon dioxide, nitrous oxide, and molten, radioactive, and nonradioactive particulates. The off-gas system must capture and processes these materials to prevent hazardous materials from entering the environment. Pumping excessive or chemically incompatible feed to the melter can cause large pressure surges, which could result in failure of the system and potentially large accidental releases.[175]

**Process Controls.** The Waste Treatment Plant will have to rely upon a complex set of engineering, administrative and operational controls, including a computer-based system that would control all aspects of facility and process operations. Understanding potential radiological, flammable, chemical, and explosive hazards and the efforts to mitigate these hazards requires accurate characterization of the chemical compositions and radionuclide concentrations at each stage of the Hanford Waste Treatment Plant process.[176] For instance, knowledge of particle size distribution and particle density of wastes, essential to design waste transfer systems such as pipes and pumps to prevent plugging, flammable gas buildup, equipment failures and accidents, remains elusive at Hanford.[177,178] Hydrogen explosions in nuclear facility piping is not an abstract issue, as there have been two hydrogen explosions in boiling water primary system pipes and an additional 25 hydrogen fires in reactor facilities and reactor pumps.[179]

**Secondary Wastes.** The Waste Treatment Plant will generate a considerable volume and high concentrations of wastes from sludge washing, ion exchange, and other processes.[180] In September 2003, DOE's contractor reported that analysis "shows significant impact from secondary wastes and thermal processes."[181]

Failed melters and related equipment are of particular concern because they are likely to contain large, irremovable concentrations of high-level

wastes.[182,183] DOE plans to dispose of failed melters in an onsite trench[184] even though Hanford "currently does not have the capability to . . . dispose of failed, highly contaminated processing equipment."[185] DOE researchers advise that, "it is unacceptable to place this waste form in relatively uncontrolled long-term storage and to continue to add more of the same and other equipment. . . ."[186]

**Bulk Vitrification.** Bulk vitrification is a supplemental treatment technology which is expected to process 60 to 70 percent of single-shell tank wastes. It involves the superheating of wastes mixed with soils containing glass-forming materials (i.e., silica, sand) with large electrodes in a large metal container. When the wastes are glassified the electrodes (melter) remain embedded in the glassified material and are disposed with the waste. Numerous bulk vitrification containers are planned, with a test project using wastes from a Hanford tank scheduled for next year. Like the melters in the Waste Treatment Plant, the success and safety of bulk vitrification will be very dependant on pretreatment and feed preparation. Several processing steps prior to vitrification have to be worked out such as: dissolution of salts for retrieval, recrystallization, chemical pretreatment, and a high-degree of moisture reduction in soil and feed. As mentioned bulk vitrification poses potentially serious long-term groundwater impacts from secondary wastes.[187] A steam explosion in 1991 at Hanford, using in situ vitrification with a 6,000 gallon tank, should serve as warning, as DOE proceed with its initial efforts to deploy this technology.

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23. U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, *Hanford Tank Waste Remediation System High-Level Waste Chemistry Manual*, Center for Nuclear Waste Regulatory Analysis, NUREG/CR-5717, 1999, p. 1–12. (Hereafter known as NUREG/CR-5717).

24. National Research Council, Board on Radioactive Waste Management, *Research Needs for High-Level Waste Stored in Tanks and Bins at U.S. Department of Energy Sites*, 2001. National Academies Press, Washington, D.C, p. 18.

25. National Research Council, Board on Radioactive Waste Management, *Nuclear Wastes: Technologies for Separations and Transmutation* (1996) National Academies Press, Washington, D.C (Hereafter known as NAS Technologies). P 89.

26. NAS Technologies p. 92.

27. B. M. Hanlon, Waste Tank Summary Report for Month Ending December 31, 2002, HNF-EP-0182, Rev. 177. CH2MHILL Hanford Group, Richland, WA.

28. K. A. Gasper, Environmental Management Science Program Workshop Hanford Tank Farm Overview, CH2MHill Hanford Group, May 6, 2003. p. 3.

29. Ibid.

30. TWINS Data 2003.

31. Defense Nuclear Facilities Safety Board, March 30, 2001 TO: K. Fortenberry, Technical Director, FROM: D. Grover and M. Satan, Hanford Site Representatives, SUBJ: Activity Report for the Week Ending March 30, 2001 (Hereafter known as DNFSB Staff Report) According to this report: "Corrosion has reduced the thickness on the interior side of the primary liner as much as 19.4 percent at a corresponding to a former waste level. The waste was out of specification for years at that level. The current waste level is below this band. The actual thinning may be substantially larger since there was extensive pitting on parts of the annulus side of the primary liner and this pit depth was not quantified by this analysis."

32. Defense Nuclear Facility Safety Board, Letter to: Paul M. Golan, Acting Assistant Secretary for Environmental Management, U.S. Department of Energy, From: John Conway, Chairman, January 18, 2005. According to the Letter: "In a recent report by a panel sponsored by the Hanford tank farms contractor and composed of nationally known chemistry and corrosion experts, Expert Panel Workshop for Hanford Site Double-Shell Tank Waste Chemistry Optimization, RPP-RPT-22126, it was stated,'... due to the paucity and fragmentary nature of the available relevant DST corrosion data, it is not currently possible to provide a clear technical basis for DST waste chemistry controls. . . .' Without a clear technical basis for DST corrosion control, changes or exemptions to the technical safety requirements (TSRs) introduce a high degree of uncertainty. The Expert Panel did endorse operating outside established chemistry control limits contingent upon the successful completion of its recommendations. The Board agrees with the Expert Panel's conclusions and recommendations."

33. National Research Council, Board on Radioactive Waste Management, *Science and Technology for Environmental Cleanup at Hanford*, Academies Press, Washington, D.C.

2001, Table 2.2 on p. 22. Total radioactivity discharged is estimated between 65,000 to 4.7 million curies.

34. NUREG/CR-5751, p. 1–9 "Records on the contents and volumes of wastes transferred to tanks are typically incomplete or nonexistent."

35. NUREG/CR-5751, p. 3-2.

36. NUREG/CR-5751, p. 3-6.

37. Vadose Zone Characterization Project at the Hanford Tank Farms, Tank Summary Data Reports for Tank BY-105, Prepared by the U.S. Department of Energy, Grand Junction Office, Grand Junction, Colorado., GJ-HAN-22, March 1996 http://www.gjo.doe.gov/programs/hanf/BYReport/bytsdr/By-105/report/content.htm, 20 July 2004 (Hereafter known as Tank Farm Summary Data Report.).

38. Tank Farm Summary Data Report for Tanks TX116 (95 tons) & 117 (41 tons), SX-133 (41 tons), TY-106 (30 tons), U-104 (60 tons) http://www.gjo.doe.gov/programs/hanf/TXREPORT/REPORT/tsdr.html, 20 July 2004.

39. DOE-ORP 2002–03, p. 10.18.

40. Thomas W. Lippman, Danger of Explosion at Nuclear Plant Covered Up, Energy Department Probe Says, *Washington Post*, August, 1, 1990.

41. Safety issues included ferrocyanide ignition, high heat, nuclear criticality, flammable organic materials, and flammable gasses.

42. U.S. Department of Energy, Press Release, Final 24 tanks removed from watch list, August 27, 2001.

43. Overview and Summary of NRC Involvement with DOE in the Tank Waste Remediation System-Privatization (TWRS-P) Program, NUREG-1747 June 29, 2001. p. 249 (Hereafter known as NUREG-1747).

44. NUREG-1747, p. 253.

45. R. E. Gephart and R. E. Lundgren. Hanford Tank Cleanup: A Guide to Understanding the Technical Issues. PNNL-10773. Richland, WA: Pacific Northwest National Laboratory. 1997.

46. U.S. Department of Energy, Defense Nuclear Facility Safety Board (DNFSB) Hanford Staff Reports, February 15, 2002, August 2, 2002, October 25, 2002. (Hereafter known as DNFSB Staff Report). According to these weekly reports, during waste retrieval, flammable gasses trapped in the salts, such as hydrogen, methane, and ammonia are released into the tank headspace. Since the initiation of the accelerated cleanup program in early 2002, several tanks undergoing waste transfers have experienced gas releases above the lower explosive limits of 25 percent, requiring work stoppage.

47. NUREG/CR-5717, p. xvii.

48. DNFSB Staff Report October 10, 2003.

49. NAS 1996, p 94.

50. 42 U.S.C. 10114 (d) "The Commission decision approving the first such application shall prohibit the emplacement in the first repository of a quantity of spent fuel containing in excess of 70,000 metric tons of heavy metal *or a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel until such time as a second repository is in operation.*" [Emphasis added]

51. NUREG 1747, p. 1. Table 2, pp. 1–3. "DOE uses the term LAW to denote. Low Activity Waste .... LAW is predominantly a liquid phase with soluble species such as

nitrates and cesium; it may also contain up to 2 percent suspended solids or solids otherwise entrained by the waste transfers. Three envelopes of LAW have been defined: Envelope A is standard, Envelope B contains higher levels of cesium, and Envelope C contains higher levels of strontium and TRU . . . . LAW would come from the liquid phases of the DSTs and from solids washing operations. From a regulatory perspective, LAW is still HLW and has high radiation levels requiring handling within shielded structures. DOE identifies the solid phases as HLW, defined as Envelope D . . . . Envelope D contains cesium, strontium, and TRUs as the radionuclides. Metal oxides, hydroxides, nitrates, phosphates, and aluminates constitute the bulk of the chemical species."

52. U.S. Department of Energy, Technical Basis for Classification of Low-Activity Waste Fraction from Hanford Site Tanks Westinghouse Hanford Corporation, WHC-SD-WM-TI-699, September 1996. (Hereafter known as WHC-SD-WM-TI-699.)

53. U.S. Nuclear Regulatory Commission, Letter to: Mr. Jackson Kinzer, Office of Tank Remediation System, U.S. Department of Energy, Richland Operations Office, From: Carl J. Paperiello, Director, Office of Nuclear Material Safety and Safeguards, June 9, 1997.

54. U.S. Department of Energy, Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada DOE/EIS-0250, February 2002, Appendix A, Table A-28. (Hereafter known as DO/EIS-0250.) DOE estimates that the total amount of chemicals in HLW glass forms at Hanford is 44,000 Kgs., compared to the total 15.1 million kilograms currently estimated in storage tanks.

55. DOE/EIS-0250, February 2002, Appendix A, p. A-39.

56. Statement of Jurisdiction, U.S. Ninth Circuit Court of Appeals, Re: *NRDC v. Abraham*, 244 F.3d 742 (9th Cir. 2003), January 29, 2004, p. 40.

57. DOE/EIS-0250, February 2002, Appendix A, Tables A-8, A-20 and A-27, at pages A-16, and A-40, respectively. Based on these tables, DOE has determined that approximately 63,000 metric tons of the total capacity of Yucca Mountain should be reserved for commercial spent nuclear fuel. The remaining 7,000 metric tons (or 10 percent) of the repository's capacity would be available for the disposal of 2,333 metric tons of DOE spent nuclear fuel and approximately 8,315 canisters (4,667 metric tons) containing solidified high-level radioactive waste from all defense sources. DOE estimates, however, that if all commercial nuclear facilities licenses are extended for 10 years, by 2046 there will be in existence approximately 105,000 metric tons of commercial spent nuclear fuel, and that by 2035 there will be approximately 2,500 metric tons of DOE spent nuclear fuel and 22,280 canisters of DOE HLW.

58. DOE/EIS-0250 February 2002, Appendix A., p. 40.

59. U.S. Department of Energy, an Evaluation of Commercial Repository Capacity for the Disposal of Defense High-Level Waste, DOE/DP0021/1 June 1985.

60. U.S. Nuclear Regulatory Commission. Memorandum for Joseph Browning, Director, Division of Waste Management, From: Neil J. Numarck, Policy and Program Control Branch, Subject: Clarification of DOE-Richland Comments on Defense Waste Commingling Study, March 14, 1985.

61. Ibid.

62. Ibid.

63. Arjun Makhijani, and Michelle Boyd, Nuclear Dumps by the Riverside: Threats to the Savannah River from Contamination at the Savannah River Site, Institute for Energy and Environmental Research, March 11, 2001, Takoma Park MD, p. 22.

64. Mathew Wald, An Effort on Atomic Waste is Called a Failure, *New York Times*, March 11, 2004.

65. National Research Council, Board on Radioactive Waste Management, *Alternative High-Level Waste Treatments at the Idaho National Engineering and Environmental Laboratory*, National Academy Press, Washington, D.C. pp. 85–86. "Another Possible conversion could be based on radioactivity measured in curies, using the fact that 1 MTHM of SBF with a burnup of 30,000 megawatt-days contains approximately 300,000 curies (Ci) after 10 years of cooling. As a result. 0.5MTHM correspond to approximately 150,000 Ci of HLW (rather than to one Savannah River-size canister). Still another conversion could be based on radiotoxicity, using regulatory release limits in 10CFR part 20 to compare the long-term performance of commercial SNF to DOE HLW based on the long-lived radionuclides in each contribute to the radiotoxicity after 1 to 10 millennia."

66. DOE/EIS-0250, February 2002, p. 8.6

67. NAS Technologies. P 94.

68. NUREG-1747, p xi. "The melters present several issues, due to their size, capacities, and surface area fluxes, all of which would make the LAW melters become the largest for radwaste vitrification in the world.

69. NUREG-1747, Table 4. pp.55–57.

70. NUREG-1747, p. 26.

71. NUREG/CR-5717, p. 1–18.

72. NUREG-1747, p. 199. The NRC areas of involvement included: Organization and General Information, Safety and Regulatory Activities, Inspections, Seismic and Structural Considerations, Hazards and Safety Analyses, Standards Approval Package (Safety Requirements Document and Hazardous Analysis Report) and Initial Safety Analysis Report, Radiation Safety and Dose Assessment Methodology Standard Review Plan, Criticality Safety Process and Chemical Safety, Standards Approval Package (Safety Requirements Document and Hazard Analysis Report) and Initial Safety Analysis, Fire Protection, Explosion Protection Issues, Environmental Protection, Standards Approval Package (Safety Requirements Document and Hazard Analysis Report) and Initial Safety Analysis Report, Quality Assurance.

73. U.S. Department of Energy, Directives, Office of Management Communication, http://www.directives.doe.gov/directives/current.html

74. Pub. L. 100-456. September 29, 1988 and United States Code, Section 2286. The responsibilities of the DNFSB are to review and evaluate standards, conduct investigations, analyze design and operational data, review facility design and construction, and make recommendations to the Secretary of Energy, authority to conduct hearings, establish reporting requirements for the Secretary of Energy, and assign resident inspectors at DOE defense nuclear facilities.

75. John Stang, Energy Department Ends BNFL Contract, *Tri-City Herald*, May 9, 2000. The privatization effort unraveled when BNFL presented a cost estimate that escalated to $15.2 billion for Phase I, instead of the expected $6.9 billion.

76. NUREG-1747, p. xi.

77. NUREG-1747, p. x.

78. NUREG-1747, p. 26.

79. 10 CFR 830.204, "A documented analysis of the extent to which a nuclear facility can be operated safely with respect to workers, the public, and the environment, including a description of the conditions, safe boundaries, and hazard controls that provide the basis for ensuring safety."

80. U.S. Department of Energy, University of California National Laboratory, Lawrence Livermore Hazard Analysis Procedure for Hazard Category 2 and 3 Nuclear Facilities Revision 1 September 2002, p. 3-2.

81. NUREG 1747, p. 169.

82. NUREG-1747, p. 50.

83. Assuming that the chances of an accident are 2.4E-02 per year, then the chances of NOT having an accident each year is 1.0 minus that value, or (1-2.4E-02). The chances of NOT having an accident at the end of two years would be (1-2.4E-02) × (1-2.4E-02) or (1-2.4E-02) ^2. Likewise, the chances of NOT having an accident at the end of N years would be (1-2.4E-02) ^N. Using N = 28, (1-2.4E-02) ^28 = 0.506518. This equates to essentially a 50 percent of NOT having an accident within 28 years.

84. NUREG-1747, p. 51.

85. NUREG-1747, p. xi.

86. NUREG-1747, Table 4.

87. NUREG-1747, p. 172.

88. NUREG-1747, p. x i. "The melters present several issues, due to their size, capacities, and surface area fluxes, all of which would become the largest for radwaste vitrification in the world. However, the experiential base, particularly from the perspective of potential ES&H concerns, is limited . . . The melter designs also have several unique attributes, including a thin gap between the cooling coils and the outer steel casing, and drainage holes. More information and analyses would be required to ascertain the safety ramifications if these melter designs are used by the new contractors."

89. NUREG 1747, Table 5 p. 60.

90. NUREG 1747, p. 60.

91. Ibid.

92. U.S. Department of Energy, Technical Evaluation of the In Situ Vitrification Melt Expulsion at the Oak Ridge National laboratory on April 21, 1996, Oak Ridge Tennessee, ORNL/ER-377.

93. U.S. Department of Energy, Environmental Health and Safety Independent Investigation of the In Situ Vitrification Melt Expulsion at the Oak Ridge National Laboratory, Oak Ridge, Tennessee, ORNL/ER-371, August 1996, p. 7.

94. Ibid.

95. NUREG-1747, p. 54.

96. NUREG-1747, p. 134–135.

97. IMAP, p. 8.4.

98. DNFSB, Staff Issue Report, Design Basis Earthquake Ground Motion Criteria for theHanford Site and Waste Treatment Plant, July 16, 2004.

99. Annette Cary, Vit plant needs more quake protection, *Tri-City Herald*, February 25, 2005. P. A-1.

100. NUREG-1747, p. 103.

101. NUREG-1747, p. 183.

102. Memorandum for Roy Schepens, Manager, Office of River Protection, From: Sandra Johnson, Director for the Office of Safety and Engineering, Subject Fire Protection for

Waste Treatment Plant, April 17, 2003. (Hereafter known as "Memo to Schepens from Johnson.")

103. Ibid.

104. Ibid.

105. DNFSB Staff Report, January 16, 2004.

106. U.S. Department of Energy, Office of Environmental Management, FY 2002 Integrated Technology Plan for the River Protection Project DOE/ORP-2002-03, p. 5.22 (Hereafter known as DOE/ORP-2002-03) Just prior to these problems emerging, DOE researcher warned: "The release of toxic gases, including the release rate and control mechanisms, must be better understood to develop compliance strategies for applicable environmental, safety, and health regulatory requirements .... No specific activity is currently under way to address this newly identified need.... As an example, construction activities associated with Tank 241-C-106 required pit access to install sluicing hardware and other equipment. The dose rate experienced in the 241-C-106 pits was 40 R/hr. After investing $2 million and 5 months, worker dose had been reduced to only 20 R/hr."

107. DOE/ORP-2002-03, 4.15.

108. DNFSB Staff Report July 11, 2003.

109. Clare Gilbert and Tom Carpenter, Knowing Endangerment: Worker Exposure to Toxic vapors at the Hanford Tank Farms, Government Accountability Project, September 2003, Washington, D.C.

110. Matthew L. Wald and Sara Kershaw, Wider Investigation Sought at Nuclear Site, *New York Times*, February 26, 2004. P. A-16.

111. Blaine Harden, Waste Cleanup May Have Human Price, *Washington Post*, March 6, 2004. P. A-1.

112. DNFSB Staff Report, February 4, 2005.

113. U.S. Department of Energy, Integrated Mission Acceleration Plan, CH2MHill, RPP 13678, Rev. 0, March 2003, (Hereafter known as IMAP). Specific technologies under active consideration include: **Steam Reforming**—"Steam reforming processes waste in a high-temperature fluidized bed under a slight vacuum. Superheated steam and additives are injected into the bed creating reducing and oxidizing zones." The process is expected to destroy organics, nitrates and nitrites. Additives are expected to incorporate radionuclides, sulfate, chlorine and fluorine into a granular waste form; and **Grout**—Grouting involves mixing wastes with compounds and conditioners such as Portland cement, fly ash, and slag to produce a cement-like waste form. In order to achieve regulatory requirements for low-level waste disposal, the dilution of radionuclides can significantly increase waste disposal volumes. Based on further performance assessments, it appears that bulk vitrification is being selected as the preferred "supplemental" technology.

114. U.S Department of Energy, Order 435.1.

115. *NRDC v. Abraham*, 244 F.3d 742 (9th Cir. 2003).

116. *The Hanford Reach*, A Publication of the U.S. Department of Energy's Richland Operations Office for all site employees, August 16, 1999, p. 7.

117. IMAP, pp. 4–18, 19

118. IMAP, p. ES-2.

119. M. Elmore and C. Henderson, *Summary of High Level Waste Tank Lay-Up Activities Supporting the Tanks Focus Area, Fiscal Years 2001–2002*, Pacific Northwest

National Laboratory, PNNL-13901, May 2002, Interim closure involves "placing tanks that no longer contain any retrievable waste and are considered to be into a safe, stable, and minimum maintenance condition until final closure could occur. This state of pre-final closure (otherwise known as interim closure, operational closure, etc.) was termed tank lay-up."

120. DOE/ORP-2002-03, p. 10.18) "Notably, borehole logging in SX Tank Farm revealed 137Cs at depths of 40 meters (130 ft), significantly deeper than predicted by current models. Further investigations, including the drilling of two additional wells, confirmed the presence of migrated cesium in the formation. The report issued by the RPP Vadose Zone Expert Panel concluded that cesium migration was poorly understood and that insufficient data were available to validate migration models . . . Furthermore, the vadose zone cleanup schedule for the 200 Areas could be delayed if the mobility status of deeply distributed contaminants is unknown or inadequately characterized well in advance. For example, if it is eventually determined that retrieval of TRU-contaminated soil down to 40 m or more beneath Plutonium Finishing Plant cribs is required, the cleanup schedule could be greatly impacted by excavation and handling costs that could approach 1 billion dollars or more. Similar excavation requirements for leaking SSTs could drive the costs of cleanup higher by several orders of magnitude. The sooner this issue is resolved, the sooner more accurate technical, financial, and schedule forecasts can be made"

121. DOE/ORP-2002-03, pp. 10.4, 10.6).

122. Energy Reorganization Act of 1974, P.L. 93-438, Oct. 11, 1974, 88 Stat. 1233, (U.S.C. Title 42, Sec. 5801 et seq.).

123. The Nuclear Waste Policy Act, P.L. 97-425, Jan. 7, 1983, 96 Stat. 2201 (Title 42, Sec. 10101 et seq.).

124. Nuclear Waste Policy Amendments Act of 1987, P.L. 100-203, title V, subtitle A, Sec. 5001-5065, Dec. 22, 1987, 101 Stat. 1330-227 to 1330-255. According to NRC (NUREG-1747, p. 215.). "Under the present system, unless the NRC determines that this LAW/incidental waste is not HLW, the waste must be disposed of as HLW in a federal repository."

125. NRC 1997 Approval Letter.

126. NUREG/CR-5751, p. 1–18, "The daughters of Sr-90 and Cs-137 –Y-90 and Ba-137m respectively – are at or near a state of transient equilibrium (i.e. equal radioactivity) with their parents and should be included in the inventory."

127. WHC-SD-WM-TI-699 Rev. 2 1996) pp. ES vi, vii.

128. U.S. Nuclear Regulatory Commission, Letter to: Mr. Jackson Kinzer, Office of Tank Remediation System, U.S. Department of Energy, Richland Operations Office, From: Carl J. Paperiello, Director, Office of Nuclear Material Safety and Safeguards, June 9, 1997. (Hereafter known as NRC 1997 Approval Letter.)

129. WHC-SD-WM-TI-699.

130. Ibid.

131. IMAP.

132. IMAP, Tank wastes scheduled for onsite disposal using "supplemental technologies" are found in IMAP (Accelerated Retrieval and Interim Closure Schedule Table 4.3, and Potentially Low Curie Low-Activity Waste Tanks Table 4.6). This estimate excludes wastes in tanks C-104,106,107,S-105,106, and 112 scheduled to go to the Waste Treatment Plant, and, wastes in tanks identified as potential transuranic wastes and, thus is based on disposition of wastes in 62 SSTs.

133. Ibid.

134. DOE/EIS-0286F, Table L.1.

135. U.S. Department of Energy, "Final Hanford Site Solid (Radioactive and Hazardous) Waste Program Environmental Impact Statement Richland, Washington" (HSWEIS), (Hereafter known as DOE/EIS - 0286F), January 2004, Table 5.15, p. 5.291, Comment: Table 5.15 of the HSWEIS provides the fraction of Maximum Contaminant Levels for Tc-99 and I-129 for several ILAW disposal sites at Hanford. The fraction of I-129 MCL (1.0 pCi I-129/L) ranges from 0.3 to 1.3. The preferred location near PUREX is 0.3 MCL. The sum of fractions MCL for Tc-99 and I-129 is 0.6 to 2.4 for various Hanford sites. The HSWEIS also identifies the Waste Treatment Plant (WTP) wastes for disposal at Hanford as containing 22 Ci I-129 in the ILAW glass and 5 Ci I-129 in grouted LLW (Liquid Effluent Treatment Facility, LETF, waste) in Table L.1. The balance of the retrieved tank waste I-129 inventory is assumed to be vitrified in HLW glass.

136. 10 CFR Part 141.16 states: Sec. 141.16 Maximum contaminant levels for beta particle and photonradioactivity from man-made radionuclides in community water systems. (a) The average annual concentration of beta particle and photon radioactivity from man-made radionuclides in drinking water shall not produce an annual dose equivalent to the total body or any internal organ greater than 4 millirem/year. (b) Except for the radionuclides listed in Table A, the concentration of man-made radionuclides causing 4 mrem total body or organ dose equivalents shall be calculated on the basis of a 2 liter per day drinking water intake

137. DOE/EIS - 0286F, p. 5.291.

138. John Rennie, Editor's Commentary: The Cold Odds against Columbia, Scientific American.Com, p. 1. "Assuming that NASA's 0.7-percent-per-mission risk estimate is correct, then over 113 missions the likelihood that one shuttle will be destroyed reaches about 55 percent." http://www.sciam.com/article.cfm?articleID=000E76D3-F389-1E43-89E0809EC588EEDF, July 20, 2004.

139. NUREG-1747, p. 16. During the period of NRC involvement, the agency had 15 staff experts, with an average experience of 20 years in their respective fields, as well as NRC's federally-funded research capabilities to focus solely on the Hanford HLW management program.

140. The Monument was established in 2000 by the President of the United States and encompasses the last free-flowing 51-mile stretch of the Columbia River, which flows through Hanford. The uniquely preserved steppe shrub environment at Hanford supports numerous species of birds, mammals, reptiles, and amphibians, including some recognized as species of concern by state and federal governments. The Reach is also the last spawning habitat for some 80 percent of wild Chinook salmon in the Pacific Northwest and shelters food and cultural resources and important archaeological sites that are vital to thousands of Native American people.

141. U.S. Department of Energy, Hanford Performance Management Plan Rev. D - August 2002, Table 1, p. iv. http://www.hanford.gov/docs/rl-2002-47/rl-2002-47.pdf, July 20, 2004

142. NUREG-1747, p. 62.

143. Ibid.

144. 10 CFR 20.

145. U.S. Environmental Protection Agency, Region 10, Columbia River Basin Fish Contaminant Survey, 1996–1998, EPA 910-R-02-006. A scientific fish consumption survey was first performed which estimated that daily average consumption of fish as 389

grams per day. Then a total of 281 samples of fish and eggs were collected from 24 study sites sampled in the Columbia River Basin, where tribal people traditionally fish. The fish and eggs were analyzed for 132 chemicals of which 92 were detected. The EPA then used its risk assessment models, hazard indices to interpret the risks of measured contaminants.

146. U.S. Department of Energy, Technical Standard, A Graded Approach for Evaluating Radiation Doses to Aquatic and Terrestrial Biota, DOE-STD-1153-2002.

147. Dose to fish: 1 rad/d Corresponding activity in fish: ~2 × 10E+6 Bq/kg, Eating 389 g/d of fish with 2 × 10E+6 Bq/kg for one year results in an effective dose of 0.0828 Sv or 8.28 rems.

148. Robert H. Nelson, *From Waste to Wilderness: Maintaining Biodiversity on Nuclear Bomb-Building Sites*, Competitive Enterprise Institute, Washington D.C. (2001) pp. 17.

149. U.S. General Accounting Office, Department of Energy: Opportunities to Improve Management of Major System Acquisitions. Report to the Chairman, Committee on Governmental Affairs, U.S. Senate GAO/RCED-97-17, (1996), Washington, D.C. (Between 1980 and 1996 GAO reported that 31 of 80 DOE major system acquisition projects had been terminated prior to completion, 34 were continuing although over budget, and 15 had been completed.)

150. National Research Council, Committee to Assess the Policies and Practices of the Department of Energy to Design, Manage, and Procure Environmental Restoration, Waste Management, and Other Construction Projects, *Improving Project Management in the Department of Energy*, National Academies Press, Washington D.C (1999).

151. National Research Council, Committee for Oversight and Assessment of U.S. Department of Energy Project Management, *Progress in Improving Project Management at the Department of Energy, 2001 Assessment*, National Academies Press, Washington D.C., (2003), p. 4 (Hereafter known as NAS-DOE Project Management 2001).

152. U.S. General Accounting Office, Report to Chairman, Subcommittee on Energy and Power Committee on Commerce, House of Representatives, Department of Energy: Contract Reform Is Progressing, but Full Implementation Will Take Years, GAO/RCED-97-18, December 1996, p. 2.

153. National Research Council, Committee for Oversight and Assessment of U.S. Department of Energy Project Management. *Progress in Improving Project Management at the Department of Energy, 2002 Assessment*, National Academies Press, Washington D.C., (2003), p. 3.

154. In the melting process, soda and lime particles normally found in soft glass, are replaced with boron oxide. Because the boron oxide particles are so small, they hold the silicate particles together more closely with aluminum oxide and sodium oxide into in a much stronger glass.

155. U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Survey of Waste Solidification Process Technologies, Center for Nuclear Waste Regulatory Analyses, NUREG/CR-6666, (2001), p. 7. (Hereafter known as /CR-6666). HLW operations include the Pamela facility in Mol, Belgium, the West Valley Demonstration Project in New York (USA), the Defense Waste Processing Facility at the Savannah River Site, (SC), the Tokai Vitrification Plant in Japan, and the Mayak facility in Russia. The Pamela and Mayak facilities were closed in 1991 and 1997, respectively. Low-level and mixed low-level operations include: the M-Area vitrification plant at the Savannah River Site, and the Fernald Silo Waste Vitrification System in Ohio. The Fernald plant closed due to technological failure in 1996.

156. NUREG/CR-6666, p. 127.

157. NUREG/CR-6666, p. 136.

158. NUREG/CR-5751, P. 8-4.

159. GAO-03-593, pp. 33–35.

160. IMAP, p. 4–17.

161. Ahearne et al., p. 6.

162. Ibid.

163. DNFSB Staff Report February 20, 2004.

164. Annette Cary, Officials to Review Vit Plant Costs, *Tri-City Herald*, January 29, 2004. P. B-1.

165. DOE/ORP-2002-03, P. 5.20.

166. DOE/ORP-2002-03, P. 8.31.

167. John Ahearne et al., "High-Level Waste Melter Review Report, U.S. Department of Energy, Office of Environmental, Management, Tanks Focus Area Report, July 2001, p. 4. "... the data supporting the current 0.77 leach factor and 1.0 percent glass solubility used by the Study Team are not strongly substantiated." (Hereafter known as "Ahearne et al.")

168. United States General Accounting Office, Report to the Ranking Minority Member, Committee on Commerce, House of Representatives, Nuclear Waste: Process to Remove Radioactive Waste From Savannah RiverTanks Fails to Work GAO/RCED-99-69, April 30, 1999.

169. M. D. Boersma and J. L. Mahoney, Glass Making Technology for High-level Waste, E. I. du Pont de Nemours and Company, Savannah River Plant, Aiken, South Carolina, August 1986.

170. U.S. National Academy of Sciences, Board on Radioactive Waste Management, *Technologies for Separations and Transmutation*, Academy Press, Washington D.C. (1996), p. 89.

171. NUREG -1747, p. 16.

172. Dhanpat Rai, Pacific Northwest National Laboratory Linfeng Rao, Lawrence Berkeley National LaboratorySue B. Clark, Washington State University Nancy J. Hess, Pacific Northwest National Laboratory, Speciation, Dissolution, and Redox Reactions of Chromium Relevant to Pretreatment and Separation of High-Level Tank Wastes Project ID: 65368, U.S. D.O.E. Environmental Management Science Program, 2000.

173. NUREG-1747, p.207 Based on world-wide experience, NRC estimates that the frequency of corrosion failures in the melter area could be as high as one 1.2E-2/yr.

174. U.S. Department of Energy, Office of River Protection, Site Need Statement, RL-WT080, October 18, 2001.

175. T. Bond Calloway, Jr., Chris T. Randall, and Victor R. Buch, Characterization of Off Gas Flow Surges in the DWPF Melter (U), Westinghouse Savannah River Company Savannah River Technology Center Aiken, SC 29808 Presentation to American Institute of Chemical Engineers Spring 1999 Conference, Houston TX, March 15–19, 1999 "Large surges can cause shutdown of the melter feed, switchover to the backup off gas system and inadvertent glass pours. Very large surges (>2 inwc) are relieved to the Melt Cell through a Seal Pot spreading contamination to Melt Cell equipment. While the Melt Cell is a shielded and remotely operated area of the plant, spread of contamination is not desirable from a maintenance perspective. Melter pressure is controlled using a standard PID algorithm with some additional features that are designed to rapidly

bring the melter pressure back to a normal operating set point (−5 inwc) whenever the system is outside the normal operating range of −2 to −10 inwc."

176. U.S. Nuclear Regulatory Commission, Programmatic review of paper entitled "PRETREAT: Graphical User Interface-Based Spreadsheet Model for Hanford Tank Waste Pretreatment Processes," Center for Nuclear Waste Regulatory Analyses, February 4, 2000. Keeping track of every element of pretreatment, alone, at the WTP involves the use of a complex computer model capable of processing large amounts of data—involving as many as 20 different worksheets.

177. DOE/ORP-2002-03, p. 6.14 This remains an unresolved issue and according to DOE "despite pipeline plugging over many years at Hanford, the technologies for removing plugs are still not well developed."

178. DNFSB Staff Report,December 10, 2004. "Preliminary calculations indicate that much of the black and hot cell piping can accumulate hydrogen volumes that are tens to hundreds of times greater than what has been considered to bea *de minimus* level."

179. DNFSB Staff Report, October 29, 2004.

180. Defense Nuclear Facility Safety Board, Letter from John Conway, Chairman to: the Honorable Spencer Abraham, Secretary of Energy, March 23, 2001. (At the Savannah River Site, the failure to address proper treatment of secondary sludge wash wastes led to the clogging of waste evaporators with significant concentrations of highly enriched uranium—creating criticality safety concerns. Moreover, the inability to process these led to the reactivation of aged single-shell tanks, which subsequently leaked. For more than a year, the Defense Waste Processing Plant at SRS was generating and storing a far larger amount of waste than it was treating.)

181. Letter (with Attachments) from: E. S. Aromi, President and General Manager, CH2M HILL, to: R. J. Schepens, Manager, Office of River Protection, U.S. Department of Energy, September 12, 2003. (Hereafter known as Aromi.)

182. NUREG-1747, p. 122. The NRC pointed out that, "For the failed HLW melter, the great majority of the vitrified waste would have to be removed in order to meet near-surface disposal requirements. It was not clear how this could be accomplished for a melter without bottom drains."

183. IMAP, p. 2.5.

184. DOE/EIS-0250, February 2002, Appendix A, Table A-41, Comment: DOE estimates at least 11 failed melters from the Savannah River Site and the West Valley Demonsotration Vitrification Plant, containing 1,052 tons of radioactive materials, generaing 46 times the decay heat of a HLW canister.

185. DOE/ORP-2002-03, pp. 7.38, 7.27.

186. Ibid.

187. Aromi.

# EXHIBIT 13

http://www.epa.gov/rpdweb00/news/wipp-news.html#wippradevent

**Radiation Protection**

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**Wipp News**

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**Remember**

To view publications with the ".pdf" extension, you will need the free Adobe Acrobat Reader. See About Portable Document Format Files to learn more about PDF files and how to download the free Reader.

Below are the latest information and documents on current topics such as recertification and ongoing inspection activities:

Get e-mail updates when this information changes.

**On this page:**

**Radiological Event at the WIPP**

**EPA Response to 2014 Radioactive Release at the Waste Isolation Pilot Plant (WIPP)**

EPA is in daily contact with DOE, the New Mexico Environment Department and the Carlsbad Environmental Monitoring and Research Center. To date, EPA's review of the data collected indicates:

- That the radiation releases do not pose public health concern.
- That DOE followed the procedures previously approved by EPA.
- That the WIPP facility remains in compliance with EPA regulations.

**Updates**

**April 14, 2014:** On Friday, April 11, EPA's WIPP team toured the WIPP Waste Handling Facility alongside DOE staff. The Agency's field team also met with the environmental sampling and radiation control groups to ensure continued coordination in all ongoing activities.

**April 11, 2014:** Yesterday, EPA's WIPP assessment and monitoring team met with DOE WIPP laboratory personnel and observed laboratory operations. Members of the EPA WIPP team also attended and presented at the Carlsbad Townhall meeting.

**April 9, 2014:** Since arriving at WIPP on Monday, members of EPA's environmental monitoring and assessment team completed an initial review of the operations of DOE's air samplers and placed EPA air samplers at three sites near existing DOE samplers.

**April 7, 2014:** Members of EPA's environmental monitoring and assessment team are arriving at WIPP today. This site visit will support EPA's larger, long-term evaluation of WIPP's continued compliance with the requirements of 40 CFR Part 191, Subpart A.

While at the WIPP facility, EPA will:

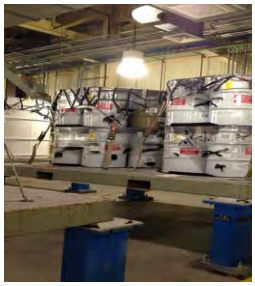
- Review DOE's air sampling plan for monitoring the area in and around the facility.
- Visit DOE's air samplers to review their operations.
- Place a limited number of air samplers on-site near existing DOE samplers to independently corroborate DOE's reported results.

**March 26, 2014:** EPA has been in daily contact with the New Mexico Environment Department (NMED) in actively overseeing the actions being taken by the Department of Energy (DOE) related to the February 2014 radiological event at the WIPP. Based on the available information, the Agency does not believe that the radioactive releases from the WIPP present public health concerns or regulatory compliance issues. As DOE plans to enter the facility, EPA will deploy several air samplers to validate and verify DOE monitoring. The Agency will continue to communicate regularly with DOE and the NMED, and will share information with the public on EPA's oversight actions at the WIPP facility through this website and other mechanisms. A one-page summary of EPA's current response and actions related to the event can also be viewed/downloaded (in Adobe PDF format) below: EPA Actions in Response to Release of Radioactive Material from the Waste Isolation Pilot Plant (PDF) (1 p, 98 K About PDF)

**WIPP Response Photo Gallery**

April 11, 2014: EPA's WIPP team tours the WIPP Waste Handling Facility with DOE staff.



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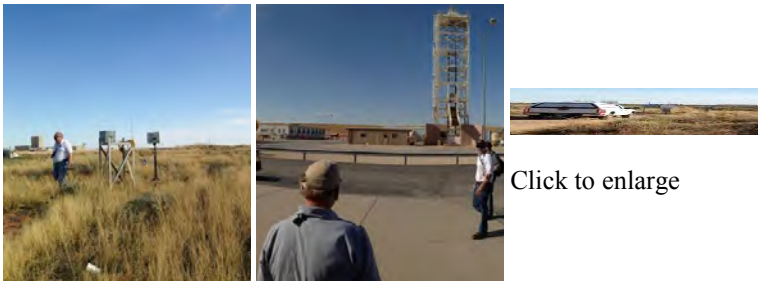
April 10, 2014: EPA staff meet with DOE staff at the WIPP laboratory. EPA WIPP team attend and present at the Carlsbad Townhall meeting.

April 9, 2014: EPA air samplers (boxes on stands) and DOE air samplers (that look like little houses) are collocated at sites near WIPP. EPA staff view the WIPP salt shaft.



Click to enlarge

April 8, 2014: EPA's WIPP team visits Department of Energy's low-volume air samplers as part of their initial review of DOE's air sampling plans.

April 7, 2014: EPA staff set up air samplers near the Waste Isolation Pilot Plant. The results from the air filter analyses will help independently corroborate Department of Energy's results.

**EPA's Role at WIPP**

EPA's role at the WIPP is to certify Department of Energy (DOE) compliance with public radiation dose and groundwater protection standards. EPA determines if DOE is in compliance by conducting regular inspections of the WIPP facility and by reviewing DOE documents and operation plans.

Learn more about EPA's oversight role at WIPP.

**Relevant Links**

- U.S. Department of Energy, Waste Isolation Pilot Plant Recovery - This site provides information about DOE's WIPP facility and its response to two isolated events took place at WIPP in February, a fire involving a salt haul truck and an alarm from a continuous air monitor (CAM) during the night shift.
- New Mexico Environment Department- Waste Isolation Pilot Plant Issue Page – This site describes NMED's response to the February 2014 Underground Salt Truck Fire and Radionuclide Release Events.
- Carlsbad Environmental Monitoring & Research Center – This site provides links to the WIPP air sampling results taken by the Carlsbad Environmental Monitoring & Research Center
- EPA Actions in Response to Release of Radioactive Material from the Waste Isolation Pilot Plant (PDF) (1 pg, 98 K, About PDF) This document describes EPA's actions in response to the February 14, 2014 release of radioactive materials from the Waste Isolation Pilot Plant.

# EXHIBIT 14

http://www.epa.gov/radiation/wipp/background.html

**Radiation Protection**

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**Waste Management:**

**About EPA's Waste Isolation Pilot Plant (WIPP) Oversight**
**WIPP**

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- About EPA's WIPP Oversight
- EPA's Role
- Inspections
- Decision Documents
- 1998 Certification
- 2006 Recertification
- 2009 Compliance Recertification Application
- Dockets
  - A-93-02
  - A-98-49
- WIPP Oversight News
- Publications
- Related Links

This page describes the history and legal authority behind EPA's role at WIPP.
**On this page:**



**EPA Response to 2014 Radioactive Release at the Waste Isolation Pilot Plant (WIPP)**

EPA is in daily contact with DOE, the New Mexico Environment Department and the Carlsbad Environmental Monitoring and Research Center. To date, EPA's review of the data collected indicates:

- That the radiation releases do not pose public health concern.

- That DOE followed the procedures previously approved by EPA.
- That the WIPP facility remains in compliance with EPA regulations.

**Update**

**April 14, 2014:** On Friday, April 11, EPA's WIPP team toured the WIPP Waste Handling Facility alongside DOE staff. The Agency's field team also met with the environmental sampling and radiation control groups to ensure continued coordination in all ongoing activities.
Learn more about EPA's response to the 2014 Radioactive Release at WIPP.

**Additional Resources**

- U.S. Department of Energy, Waste Isolation Pilot Plant Recovery
- New Mexico Environment Department- Waste Isolation Pilot Plant Issue Page
- Carlsbad Environmental Monitoring & Research Center
- EPA Actions in Response to Release of Radioactive Material from the Waste Isolation Pilot Plant (PDF)(1 pg, 98 K, About PDF)
- See EPA's WIPP Response Photo Gallery

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**EPA's Role at WIPP**

In the Waste Isolation Pilot Plant (WIPP) Land Withdrawal Act, Congress required the U.S. Environmental Protection Agency (EPA) to issue final regulations regarding the disposal of spent nuclear fuel, high-level radioactive waste, and transuranic waste. It also gave EPA the authority to develop the criteria that implement the final radioactive waste disposal standards specifically for the WIPP. In addition, EPA must determine whether WIPP may be re-certified every five years until the facility is decommissioned. Finally, the WIPP LWA required EPA to determine that the WIPP complies with other federal environmental and public health and safety regulations, such as the Clean Air Act and the Solid Waste Disposal Act. You can find additional information on EPA's WIPP role on "EPA's Continuing Role."
The WIPP is the nation's first facility for deep geological disposal of transuranic radioactive waste (TRU). It has been developed by the U.S. Department of Energy in southeastern New Mexico, about 26 miles east of Carlsbad. Public Law 102-579, also called the Waste Isolation Pilot Plant Land Withdrawal Act (WIPP LWA) withdrew an area of 10,240 acres from public use in October 1992.
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**Final radioactive waste disposal standards**

On December 3, 1993, EPA issued final amendments to its radioactive waste disposal standards, which were initially promulgated in 1985 (40 CFR Part 191). The amendments address the individual and ground water protection requirements of the original standards, which had been remanded by the U.S. Court of Appeals. The other portions of the standards were not amended.
The individual protection requirements were amended to require disposal systems to be designed to limit the amount of radiation to which an individual can be exposed for 10,000 years, rather than for 1,000 years as was required in the original standard. The final ground water protection requirements were amended to require disposal systems to be designed so that, for 10,000 years after waste disposal, contamination in off-site underground sources of drinking water will not exceed the maximum contaminant level for radionuclides established by the EPA under the Safe Drinking Water Act.
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**Compliance Criteria for the WIPP**

On February 9, 1996, EPA issued final compliance criteria (40 CFR Part 194) for the certification and re-certification of the WIPP's compliance with the final radioactive waste disposal standards. The compliance criteria are divided into four subparts:

Subpart A contains definitions of terms, references, and reporting requirements for DOE. It also describes EPA's authority to modify, suspend, or revoke certification or re-certification.

Subpart B describes the procedure for submission of any compliance application, and specifies the content of applications.

Subpart C consists of requirements that apply to activities undertaken to demonstrate compliance with EPA's disposal standards. General requirements pertain to quality assurance, the use of computer models to simulate the WIPP's performance, and other areas. Containment requirements limit releases of radionuclides to specified levels for 10,000 years after the facility accepts its final waste for disposal. Assurance requirements involve additional measures intended to provide confidence in the long-term containment of radioactive waste. Also, Subpart C implements requirements in the disposal standards for protecting individuals and ground water from exposure to radioactive contamination.

Subpart D describes the process for public participation that EPA will follow for certification and re-certification decisions.

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**The 1998 Certification Decision**

DOE submitted a Compliance Certification Application (CCA) to EPA on October 29, 1996, to demonstrate that the WIPP complies with the criteria at 40 CFR Part 194. After receiving the CCA, EPA published an Advance Notice of Proposed Rulemaking in the Federal Register that announced receipt of the application and initiated a 120-day public comment period. Copies of the application were made available for public inspection at the Agency WIPP docket and the three supplemental New Mexico Dockets. Written comments were solicited, and public hearings were held in New Mexico in February 1997.

Over the next several months, EPA requested additional information from DOE related to the completeness and technical sufficiency of the CCA. EPA announced its finding that the CCA was complete in the Federal Register on May 22, 1997.

EPA published a Notice of Proposed Rulemaking in the Federal Register on October 30, 1997, announcing the proposed certification that the WIPP will comply with EPA's disposal standards. The proposed decision is accompanied by Compliance Application Review Documents (CARDs) that further explain the technical basis for EPA's decision and contain EPA's responses to comments received on the Advance Notice of Proposed Rulemaking. The announcement of the proposal initiated a 120-day period in which the public commented on the Proposed Certification. During this comment period the Agency held public hearings in Carlsbad, Albuquerque, and Santa Fe, New Mexico. EPA's Final Rulemaking Notice on the certification decision was announced on May 13, 1998. This decision is accompanied by a document that summarizes significant comments and issues regarding the proposal and provides EPA's responses.

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**EPA's Recertification Decisions**

EPA is required by law to evaluate all changes in conditions or activities at WIPP every five years to determine if the facility continues to comply with EPA's disposal regulations. The Agency has undergone two of these "recertification" processes—initially in 2004 and again in 2009—which included a review of all of the changes made at the WIPP facility since the original 1998 EPA certification. Recertification is not a reconsideration of the decision to open WIPP, but a process to reaffirm that WIPP meets all requirements of the disposal regulations. The recertification process is not used to approve any new significant changes proposed by DOE; any such proposals will be addressed separately by EPA. Recertification ensures that WIPP's continued compliance is demonstrated using the most accurate, up-to-date information available.

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**Other Regulatory Agents**

**EPA**

The Office of Radiation and Indoor Air (ORIA) coordinates most of EPA's actions under the WIPP LWA. However, other EPA offices also play important roles in the regulation of WIPP. EPA's Region VI office, based in Dallas, Texas, is responsible for determining the WIPP's compliance with all applicable environmental laws and regulations other than the radioactive waste disposal standards.

The Region VI office also coordinates with the EPA Office of Solid Waste on hazardous waste issues. Some transuranic radioactive waste intended for disposal at the WIPP also contains hazardous components, thus subjecting it to the regulations developed under the Resource Conservation and Recovery Act.

- EPA Mixed Waste Home Page

**State of New Mexico**

In addition, the State of New Mexico is authorized by EPA to carry out the State's base RCRA and mixed waste programs in lieu of the equivalent Federal programs. New Mexico's Environment Department reviews permit applications for treatment, storage, and disposal facilities for hazardous waste, under Subtitle C of RCRA.

- New Mexico Environment Department Home Page EXIT Disclaimer

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# EXHIBIT 15

http://www.wipp.energy.gov/wipprecovery/recovery.html

Last Updated: 4/13/14



**WIPP Update**
**April 13, 2014**
**There is no new WIPP Update today. Please see below for the most recent information. The next WIPP Update will be issued on Monday, April 14.**

**April 12, 2014**
**Carlsbad Environmental Monitoring and Research Center:**
**WIPP contamination below minimum detectable concentrations**

The Carlsbad Environmental Monitoring and Research Center (CEMRC), an independent scientific and research group associated with New Mexico State University, released new WIPP monitoring data yesterday that reveals no contamination in surface water samples collected from three area reservoirs. CEMRC's testing shows that the areas previously showing trace amounts of radioactive particles following the February 14 event have returned to historic background levels. This confirms results from WIPP's environmental monitoring program. All recent sampling and analysis conducted in the area by WIPP employees do not detect contamination in the area's air, surface water, sediment, vegetation, soil, or wildlife.

Available sampling data is posted on the WIPP website.

**Community Meetings Scheduled**

April 14 – WIPP update to the Jal City Council at 5:30 p.m. Location: 309 S. Main St.

April 17 – Carlsbad Mayor Dale Janway and DOE will co-host a weekly town hall meeting featuring updates on WIPP recovery activities. The meetings are held every Thursday at 5:30 p.m. Location: Carlsbad City Council Chambers, 101 N. Halagueno Street. Live streaming of the weekly meetings can be seen at http://new.livestream.com/rrv/.

NOTE: There will not be a WIPP Update for Sunday, April 13. The next update will be Monday, April 14.

**WIPP Recovery Checklist**


WIPP has developed a checklist as part of the recovery process. Click the image above to view these steps.

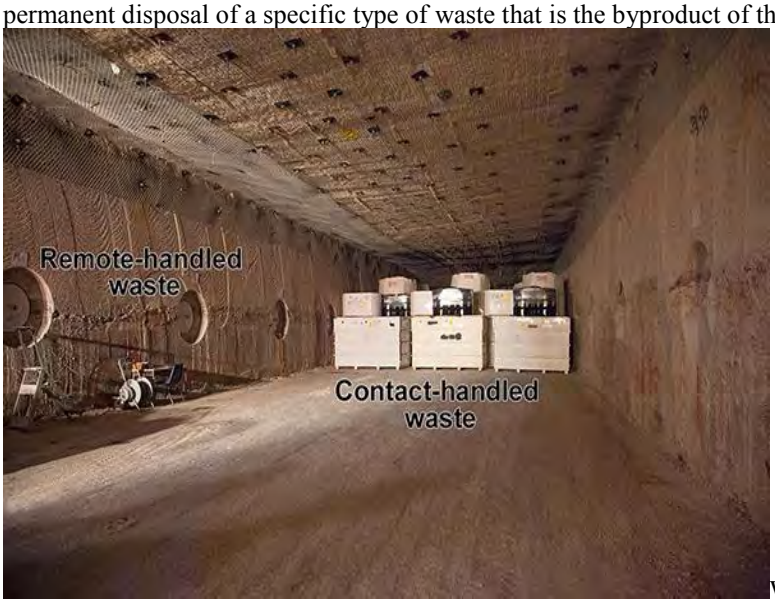
***Did you know?***

The salt formation containing the WIPP repository is about 2,000 feet thick, beginning 850 feet below the surface. For more information about WIPP, see our Fact Sheets.

**About WIPP**

***The nation's only deep geologic repository for nuclear waste***

The U.S. Department of Energy's (DOE) Waste Isolation Pilot Plant (WIPP) is a deep geologic repository for permanent disposal of a specific type of waste that is the byproduct of the nation's nuclear defense program.




WIPP is the nation's only repository for the disposal of nuclear waste known as transuranic, or TRU, waste. It consists of clothing, tools, rags, residues, debris, soil and other items contaminated with small amounts of plutonium and other man-made radioactive elements. Disposal of transuranic waste is critical to the cleanup of Cold War nuclear production sites. Waste from DOE sites around the country is sent to WIPP for permanent disposal.

TRU waste is categorized as "contact-handled" or "remote-handled" based on the amount of radiation dose measured at the surface of the waste container. Contact-handled waste has a radiation dose rate not greater than 200 millirem (mrem) per hour, while remote-handled waste can have a dose rate up to 1,000 rem per hour. About 96 percent of the waste to be disposed at WIPP is contact-handled.

TRU waste is long-lived and has to be isolated to protect public health and the environment. Deep geologic disposal in salt beds was chosen because the salt is free of flowing water, easily mined, impermeable and geologically stable. Salt rock also naturally seals fractures and closes openings.

The WIPP site, located in southeast New Mexico about 26 miles from Carlsbad, was constructed in the 1980s for disposal of defense-generated TRU waste. The underground repository is carved out of a 2,000-foot-thick salt bed formed 250 million years ago. TRU waste is disposed of 2,150-feet underground in rooms mined from the salt bed. WIPP has been disposing of legacy TRU waste since 1999, cleaning up 22 generator sites nationwide.


**Recovery Page**
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**Event Information**

Accident Description
Protective Actions
Sampling Results
Path Forward

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**News and Updates**

Past WIPP Updates
News Releases
Fact Sheets
Photo and Video

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**Other WIPP Documents and Tools**

**Hazardous Waste Permit**

Information Repository
Stakeholder Notification
Stakeholder Documents
Submitted Modifcations/Requests for Comments
Information of Proposed Modifications

**WIPP Toolbox**

WIPP Waste Information System Public Access
CH Bay VOC Monitoring Report Page

# EXHIBIT 16

http://www.nmenv.state.nm.us/NMED/Issues/WIPP2014.html

- Air
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- Health & Safety
- Cleanups
- Permits & Licenses
- Services & Assistance

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**Waste Isolation Pilot Plant (WIPP)**
**Response to February 2014 Underground Salt Truck Fire and**
**Radionuclide Release Events**


**The Issues and Timeline:**

- February 5, 2014, a diesel powered salt-hauler vehicle caught fire in the underground forcing workers to evacuate and operations to cease. The cause of the fire has not yet been determined, however this event occurred in an area removed from where waste is handled and emplaced. Waste handling operations in the underground will cease while the event is investigated.
- February 14, 2014, at 11:13 PM a continuous air monitoring (CAM) alarm in the underground indicated the detection of radiation above background trigger points. A second alarm followed immediately indicating detection of radiation at higher levels. This triggered a switch from exhausing air to the environment to first passing exhaust air (effluent) through a filtration system before exhausting to the environment.
- Since that time, the Department of Energy (DOE) and other sources have confirmed trace amounts of particulate radiation released to the surface and into the atmosphere at the WIPP facility. The release is being monitored by DOE, NMED, and the Carlsbad Environmental Monitoring Research Center (CEMRC).
- The WIPP underground remains shutdown as the DOE and others investigate the cause of the event that released radioactive material to the underground, exhaust system, and surface.
- February 27, 2014, NMED issued an Administrative Order does not allow commencement of normal operations until NMED inspects and approves the facility (more below).
- Laboratory analyses of filter media collected following the event are being processed. Following data validation, these results are made available to the public.
- Plans are being prepared to conduct additional environmental sampling and analysis to determine the impact this event has on human health and the environment.
- 21 March, 2014: NMED withdrew WIPP's draft Permit for the Class 3

**To view all posted WIPP incident documents go to...**

**WIPP Documents Page**

PAGE CONTENTS:
Monitoring Data & Reports
Maps & photos
Regulatory Information
Background Information

**Current data and other documents will be posted as they become available.*

**UPDATES**

**3/21/14**
NMED Withdraws Draft for Permit Modification Request
**3/20/14**
DOE, NWP Making Plans to Continue Moving LANL Transuranic Waste

modification. This will also close the public comment period for this draft. (link to HWB notice and letter)

- April 2, 2014: The first team has completed surveys and established a base camp in the underground. A second team is headed underground to check conditions between the Salt Hoist and the Air Intake Shaft. The remainder of today's re-entry activities are expected to take about three hours.

**WEEKLY TOWN HALL MEETINGS:**

- **Thursdays at 5:30 - Carlsbad City Council Chambers, City Hall, 101 N. Halagueno St.**
- Co-hosted by City of Carlsbad and U.S. Department of Energy (DOE) Carlsbad Field Office
- Discuss recovery efforts following WIPP's fire and radiological events in February.
- **Meetings available live online at http://new.livestream.com/rrv.**
-

- **Who is Involved**
- **NMED's Role**
- **Administrative Order**
- **Documents & Links**
- **Regulations**

**Who is Involved:**

*Multiple parties, including those listed below, are working to investigate and monitor impacts to human health and the environment related to these events.*

- **U.S. Department Of Energy (DOE)** - WIPP is a DOE facility. [WIPP Website]
- **U.S. Environmental Protection Agency (EPA)** - The EPA's role is to ensure that WIPP protects human health and the environment during the disposal phase, the repository closure phase, and the repository's long-term performance post-closure. The EPA is responsible for certifying the facility's compliance with repository performance requirements. The original certification was granted in 1999. EPA also reviews new information from DOE to determine whether the certification should be modified, suspended or revoked and must conduct a recertification of the facility every 5 years. Visit EPA's website for more information on EPA's history and legal authority at WIPP.[EPA's Role] -and- [EPA's Activities]
- **Nuclear Waste Partnership LLC (NWP)** - The U.S. DOE contracts with NWP to manage and operate WIPP. [NWP Website]
- **New Mexico Environment Department (NMED)** - NMED's Hazardous Waste Bureau regulates the hazardous waste component of mixed waste emplaced at WIPP and provides technical guidance to all New Mexico hazardous waste generators, treatment, storage, and disposal facilities as required by the New Mexico Hazardous Waste Act. In accordance with the Act, NMED issued the original hazardous waste permit to WIPP in 1999. The permit must be renewed every 10 years. Compliance with the permit is determined by NMED.

  NMED Oversight Bureau conducts oversight and monitoring activities at WIPP. Monitoring programs include both joint and independent evaluations of impacts to the environment and public health. Environmental sampling and analysis include particulates in air, soils and sediments, vegetation, surface and ground water.

  ***New Mexico does not have authority to regulate the radiological components of the mixed waste at WIPP. Radiological emissions are regulated by EPA's Radiation Protection Program.*
- **N.M. Radioactive Waste Consultation Task Force (a.ka. Governor's WIPP Task Force)-** The membership is comprised of the Secretaries of the Energy, Minerals and Natural Resources Department, Department of Health, Environment Department, Department of Public Safety, the Department of Homeland Security and Emergency Management, the Department of Transportation, and the State Fire Marshal or their designees.
- **Carlsbad Environmental Monitoring & Research Center (CEMRC)** - a division of the College of Engineering at New Mexico State University. CEMRC facility and staff provide support to WIPP, LANL, SNL, and WTS primarily through site and environmental monitoring, *in vivo* bioassay, and scientific and laboratory support. [CEMRC website]

# EXHIBIT 17

http://www.cemrc.org/

New Mexico State University

New Mexico State University
Carlsbad Environmental Monitoring & Research Center

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**Carlsbad Environmental Monitoring & Research Center**

The Carlsbad Environmental Monitoring & Research Center is a division of the College of Engineering at New Mexico State University.

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**About us**

The Carlsbad Environmental Monitoring & Research Center is a division of the College of Engineering at New Mexico State University. This 26,000 $ft^2$ radiochemistry facility includes environmental and general radiochemistry laboratories, a special plutonium-uranium lab, an *in vivo* bioassay facility, mobile laboratories, computing operations and offices. The facility can perform a wide range of environmental and radiochemistry work, characterization, monitoring, and feasibility studies in support of performance assessment, radiological and environmental training and education, subsurface flow and transport experiments, nuclear energy issues, and issues involving Homeland Security particularly those involving radiation dispersal devices (RDDs or dirty bombs). CEMRC has partnered with Los Alamos National Laboratory (LANL), Sandia National Laboratory (SNL), and Nuclear waste Partnership, LLC (NWP) to create a unique facility with programs that include: environmental monitoring of almost any radiological and inorganic constituent; actinide chemistry and repository science particularly concerning the environmental behavior of Pu, Am, U and Np; dirty bomb mitigation research and training particularly for $^{137}Cs$ and $^{60}Co$, head space gas and volatile organic compound (VOC) analyses; *in vivo* and *in vitro* bioassay, whole body dosimetry, military small arms range clean-up, evaluation and design of innovative treatment technologies, and soil, water, air and waste characterization.

The Department of Energy Carlsbad Field Office (DOE CBFO) currently operates the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico, as a repository site for transuranic (TRU) waste generated as part of the nuclear defense research and production activities of the federal government. The CEMRC facility and staff provide support to WIPP, LANL, SNL and WTS primarily through site and environmental monitoring, *in vivo* bioassay, and scientific and laboratory support.

# EXHIBIT 18

**EPA Actions in Response to Release of Radioactive Material from the Waste Isolation Pilot Plant (WIPP)**

Since the release of radioactive material on February 14, 2014 at the WIPP, EPA has been working closely with the New Mexico Environment Department (NMED) in actively overseeing the actions being taken by the Department of Energy (DOE). Based on the available information, EPA does not believe that the radioactive releases from the WIPP present public health concerns or regulatory compliance issues. However, EPA is in daily communications with DOE and NMED and is taking the following actions to support and provide oversight of the response effort, and provide information for the public:

Review of DOE's Data and Analysis

- ***Data analysis and review*** – EPA is reviewing data provided by DOE, NMED and the independent Carlsbad Environmental Monitoring and Research Center (CEMRC) for consistency, completeness and to identify trends or unexpected results. EPA is also conducting a verification and validation of DOE-generated lab data, which will include a review of a representative portion of DOE data.
- ***Public dose and dispersion modeling*** – EPA is evaluating DOE's plume modeling and dose projections for reasonableness and rigor, and also conducting independent modeling of public dose projections using data supplied by DOE.

EPA Verification and Review of DOE Air Monitoring

- ***Independent EPA air monitoring –*** EPA's RadNet Monitor in Carlsbad continues to show radionuclide levels at background. Also, within the next 1-2 weeks, EPA plans to deploy 3-4 air samplers at the WIPP facility to validate and verify DOE monitoring. These samplers are expected to be located on-site near existing DOE samplers and will be in place for a limited period of time during DOE's re-entry into the underground. The monitors will provide extra support during re-entry as well as independent verification of DOE's monitoring network. As the incident response progresses, EPA will continue to reevaluate whether additional monitors are needed in the near term.
- ***Evaluation of DOE environmental monitoring system*** – EPA is evaluating the number and location of DOE's environmental monitors to identify any needed improvements for the future to ensure adequate monitoring is in place should a similar incident occur in the future.

Oversight at the WIPP Site

- ***On-site visits and inspections*** – EPA plans to continue to conduct site visits, as needed, to oversee DOE's management of the incident including:
  - EPA staff was on-site for an initial incident assessment and to attend public meetings;
  - A more extensive site visit is planned shortly (along with the monitoring described above) to inspect the facility and gather information for EPA's review of DOE's environmental monitoring; EPA will conduct a follow-on site visit(s) to coordinate with DOE's incident investigation and to evaluate potential process improvements;
  - A complete compliance inspection will be conducted at a later date when EPA can have full access to the underground, including the waste panels.

EPA Communications

- EPA will continue to communicate regularly with DOE and the NMED, and will share information with the public on EPA's oversight actions at the WIPP facility through EPA's website and other mechanisms.

# EXHIBIT 19

SCIENTIFIC AMERICAN™

**Radiation Levels Fall after Nuclear Waste Leak in New Mexico**

The U.S. Department of Energy is preparing to reenter the deep-underground waste repository to determine the cause of the spike

nature

Feb 26, 2014 |By Jeff Tollefson and Nature magazine

The WIPP underground facility stores drums of radioactive material from nuclear weapons labs.
*DOE WIPP*

Radiation levels within and around the United States' only deep-underground nuclear waste facility continue to drop, nearly two weeks after a mysterious leak triggered alarms and shut down the facility, according to data released this week by the US Department of Energy and an independent air-monitoring group.

The sharp spike and subsequent decline in radiation are suggestive of a single release of contamination on 14 February at the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico. It is the first reported leak at the WIPP, which is a permanent repository for nuclear waste that has been carved out of ancient salt beds 655 metres underground. Contamination escaped the facility, but officials say that the levels are low and pose no health threat.

Because no one was underground when the radiation alarms went off, it remains unclear what caused the release. One possibility is that a large chunk of salt fell from the ceiling of the repository and damaged one of the metal storage drums, says Russell Hardy, director of the Carlsbad Environmental Monitoring and Research Center at New Mexico State University, which independently monitors radiation at the site. "But until they get underground and find out what happened, it's really all just speculation at this point," he says.

The WIPP opened in 1999 and has since taken in more than 80,000 cubic metres of material – including work gloves, tools and machinery – that is contaminated with radioactive elements such as plutonium as well as hazardous chemicals. On 16 February, two days after the initial release, Hardy's centre detected plutonium and americium contamination at an air-monitoring station 1 kilometre away from an exhaust shaft leading from the facility. The centre's latest results, released on 25 February based on samples collected four days after the leak, identified no plutonium and sharply lower levels of americium. Hardy says that the centre's data align with reports from the US Department of Energy. The agency estimated that a person at one of its above-ground monitoring stations would have sustained a cumulative radiation exposure of 1 millirem – ten times less radiation than that delivered during a typical chest X-ray.

Although no data were released from real-time radiation detectors within the facility this week, the Department of Energy says that radiation levels are dropping and seem to be limited to one section of the facility. Energy Department spokeswoman Deb Gill says that the agency and its contractors – an industry consortium led by the San Francisco-based URS Corporation – are still working on a plan to re-enter the facility.

The leak came nine days after an apparently unrelated incident in which a vehicle caught fire underground. The Department of Energy had already appointed a panel to investigate the fire, and that panel will now investigate the radiation leak as well, Gill says.

Hardy says his group is still analysing samples taken directly from the exhaust shaft – both before and after the air is filtered – and plans to release the results as early as today. The findings will determine whether the air filtration

system, which is designed to capture 99.97% of the radiation, functioned properly. “We’ve never had to test it under live conditions,” Hardy says.

# EXHIBIT 20

# U.S. Department of Energy
# Office of Environmental Management

# Accident Investigation Report


## Underground Salt Haul Truck Fire at the Waste Isolation Pilot Plant February 5, 2014

**March 2014**

## Disclaimer

This report is an independent product of the Accident Investigation Board appointed by Matthew Moury, Deputy Assistant Secretary, Safety, Security, and Quality Programs, U.S. Department of Energy, Office of Environmental Management. The Board was appointed to perform an Accident Investigation and to prepare an investigation report in accordance with Department of Energy (DOE) Order 225.1B, Accident Investigations.

The discussion of the facts as determined by the Board and the views expressed in the report do not assume and are not intended to establish the existence of any duty at law on the part of the U.S. Government, its employees or agents, contractors, their employees or agents, or subcontractors at any tier, or any other party.

This report neither determines nor implies liability.

## Release Authorization

On February 7, 2014, an Accident Investigation Board was appointed to investigate a fire at the U.S. Department of Energy, Waste Isolation Pilot Plant site near Carlsbad, New Mexico, that occurred on February 5, 2014. An aged EIMCO 985-T15 dump truck (salt haul truck) caught fire in the underground. The Board's responsibilities have been completed with respect to this investigation. The analysis and the identification of the contributing causes, the root cause and the Judgments of Need resulting from this investigation were performed in accordance with DOE Order 225.1B, *Accident Investigations*.

The report of the Accident Investigation Board has been accepted and the authorization to release this report for general distribution has been granted.

MB Moury

Matthew Moury
Deputy Assistant Secretary
Safety, Security, and Quality Programs
Office of Environmental Management

3/13/14
Date

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## Figures

# Tables

## Acronyms

| | |
|---|---|
| ALARA | As Low as Reasonably Achievable |
| BNA | Baseline Needs Assessment |
| CAS | Contractor Assurance System |
| CBFO | Carlsbad Field Office |
| CC | contributing cause |
| CH | contact handled |
| CHAMPS | Computerized History and Maintenance Planning System |
| CLR | Conveyance Loading Room |
| CMC | Carlsbad Medical Center |
| CMR | Central Monitoring Room |
| CMRO | Central Monitoring Room Operator |
| CMS | Central Monitoring System |
| CMT | Crisis Management Team |
| CON | Conclusion |
| CONOPS | Conduct of Operations |
| DOE | U.S. Department of Energy |
| DC | Direct Cause |
| DNFSB | Defense Nuclear Facilities Safety Board |
| EAL | emergency action level |
| EAP | Employee Assistance Program |
| EMCBC | Office of Environmental Management Consolidated Business Center |
| EMS | Emergency Medical Services |
| EOC | Emergency Operations Center |
| EPA | U.S. Environmental Protection Agency |
| ERO | Emergency Response Organization |
| ERT | Emergency Response Team |
| EST | Emergency Services Technician |
| EXO | Enriched Xenon Observatory |
| FHA | Fire Hazard Analysis |
| FLIRT | First Line Initial Response Team |
| FPP | Fire Protection Program |
| FR | Facility Representative |
| FSM | Facility Shift Manager |
| FSS | Fire Suppression System |
| GET | General Employee Training |

| | |
|---|---|
| HEPA | high-efficiency particulate absorption |
| HQ | Headquarters |
| ICS | Incident Command System |
| ISM | Integrated Safety Management |
| ISMS | Integrated Safety Management System |
| JHA | Job Hazard Analysis |
| JON | Judgments of Need |
| JIC | Joint Information Center |
| LPU | Local Processing Unit |
| M&O | Management and Operations |
| MRT | Mine Rescue Team |
| MST | Mountain Standard Time |
| MW | Megawatt |
| NFPA | National Fire Protection Association |
| NWP | Nuclear Waste Partnership LLC |
| MSHA | Mine Safety and Health Administration |
| MST | Mountain Standard Time |
| OE | Operational Emergency |
| O&M | Operations and Maintenance |
| PA | public address |
| RH | Remote handled |
| RC | Root Cause |
| RCRA | Resource Conservation and Recovery Act |
| SAA | Shaft Access Area |
| SCFM | standard cubic feet per minute |
| SCSR | Self-Contained Self-rescuer |
| SLA | Service Level Agreement |
| SME | subject matter expert |
| SMP | Safety Management Program |
| TRU | Transuranic |
| U/G | Underground |
| WH | Waste Handling |
| WIPP | Waste Isolation Pilot Plant |

# Executive Summary

On Wednesday, February 5, 2014, at approximately 1045 Mountain Standard Time, an underground mine fire involving an EIMCO Haul Truck 74-U-006B (salt haul truck) occurred at the Department of Energy (DOE) Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico. There were 86 workers in the mine (underground) when the fire occurred. All workers were safely evacuated. Six workers were transported to the Carlsbad Medical Center (CMC) for treatment for smoke inhalation and an additional seven workers were treated on-site.

On February 7, 2014, Matthew Moury, Deputy Assistant Secretary for Safety, Security, and Quality Programs, U.S. Department of Energy, Office of Environmental Management formally appointed an Accident Investigation Board (the Board) to investigate the accident in accordance with DOE Order (O) 225.1B, based on this accident meeting Accident Investigation Criteria 2.d.1 of DOE O 225.1B, *Accident Investigations*, Appendix A.

The Board began the investigation on February 10, 2014, completed the investigation on March 8, 2014, and submitted findings to the Deputy Assistant Secretary for Safety, Security, and Quality Programs Environmental Management on March 11, 2014.

The Board concluded that this accident was preventable.

**Accident Description**

The fire is believed to have originated in the truck's engine compartment and involved hydraulic fluid and/or diesel fuel which contacted hot surfaces on the truck, possibly the catalytic converter, and then ignited. The fire burned the engine compartment and consumed the front tires which contributed significantly to the amount of smoke and soot in the underground.

The Operator had just unloaded salt from the truck at approximately 1045 Mountain Standard Time (MST) when he noticed an orange glow and then flames between the engine and the dump sections of the truck (see Figure ES-1). The Operator attempted to extinguish the fire with a portable fire extinguisher stored on the truck and then by activating the salt haul truck's fire suppression system. Both attempts to extinguish the fire were unsuccessful. The Operator then used a mine phone to notified Maintenance of the fire, and his Supervisor overheard the conversation over a nearby mine phone, which can also be heard throughout the underground. Two nearby workers heard the discussion on the mine phone and, based on the urgency of the Operator's voice, went to the scene to see if they could assist. They began pushing a nearby 300-pound fire extinguisher to the fire when their carbon monoxide monitor alarmed and the smoke worsened. One of the workers called the Central Monitoring Room (CMR) to report the fire and smoke, and recommended evacuation of the underground.


**Figure ES-1: EIMCO Haul Truck 74-U-006B after Fire**

At 1051, the Central Monitoring Room Operator (CMRO) sounded the evacuation "yelp" alarm for approximately two seconds and then made a public address system (PA) announcement that there was a fire in the underground and for all personnel to evacuate via the area egress stations. A subsequent announcement directed the workers to the waste hoist. As reported by some workers, this instruction was not heard throughout the underground. Some workers learned of the fire and need to evacuate through the "chatter" (discussions) on the mine phone, through co-workers, or through their supervisors.

At 1058, the Facility Shift Manager (FSM) directed the CMRO to switch the ventilation system from normal to filtration mode believing this would reduce both the fire and smoke in the underground. However, this resulted in the flow of smoke into areas of the underground, which the workers expected to have "good" air. The first group of workers arrived at the waste hoist and the first of three trips to evacuate the workers from the mine via the Waste Hoist (mantrips) to the surface was completed. The CMR activated the Emergency Operations Center (EOC) at 1103 and the Joint Information Center (JIC) was activated at 1125.

Other workers continued to make their way on foot or on electric carts from various locations throughout the underground to the waste hoist. At this point, there was smoke in most areas of the underground and smoke could be seen on the surface exiting the Salt Handling Shaft. Workers had difficulty reaching the waste hoist due to poor visibility from their primary evacuation routes and obscured evacuation route reflectors; this was compounded by a delay in activating the evacuation strobe lights. Some workers also had difficulty opening and/or donning their self-rescuers or self-contained self-rescuers (SCSRs). The second mantrip of underground

personnel was completed at 1120 and the third and final mantrip was completed at 1134. Full accountability of all underground workers was achieved at 1135.

All surface waste-handling activities were suspended and the Mine Rescue Team (MRT) was activated at 1120.

Once on the surface, workers were evaluated by Emergency Service Technicians (ESTs) and six personnel were transported to the CMC for treatment of smoke inhalation. At 1420, all personnel were released from the CMC.

The MRT performed carbon monoxide gas checks and entered the underground via the Air Intake Shaft at 1746. They proceeded to the reported fire location via the Air Intake Shaft and arrived at the salt haul truck at 1825. No fire was observed. Oxygen levels were at 21 percent and methane and carbon monoxide were at 0 percent. The MRT noted that the air was clear but that there were embers at the location of the right front tire. They expended their fire extinguishers on these embers and proceeded to the surface at 1915.

At 2202, a second MRT entered the underground via the salt hoist, took additional air quality readings, and drove the underground rescue vehicle to the scene of the fire. They applied all the extinguishing foam from the rescue vehicle and the fire appeared to be fully extinguished. They then unchained a number of bulkhead doors which had been chained open prior to the incident. On Thursday, February 6, 2014, at 0025, the MRT exited the underground via the salt hoist.

At 0105 on February 6, 2014, the event was terminated and the EOC and JIC were deactivated.

**Direct, Root, and Contributing Causes**

**Direct Cause** (DC) – the immediate events or conditions that caused the accident.

The Board identified the direct cause of this accident to be contact between flammable fluids (either hydraulic fluid or diesel fuel) and hot surfaces (most likely the catalytic converter) on the salt haul truck, which resulted in a fire that consumed the engine compartment and two front tires.

**Root Cause** (RC) – causal factors that, if corrected, would prevent recurrence of the same or similar accidents.

The Board identified the root cause of this accident to be the failure of Nuclear Waste Partnership LLC (NWP) and the previous management and operations (M&O) contractor to adequately recognize and mitigate the hazard regarding a fire in the underground. This includes recognition and removal of the buildup of combustibles through inspections and periodic preventative maintenance (e.g., cleaning), and the decision to deactivate the automatic onboard fire suppression system.

**Contributing Causes** (CC) – events or conditions that collectively with other causes increased the likelihood or severity of an accident but that individually did not cause the accident. For the purposes of this investigation, contributing causes include those related to the cause of the fire, as well as those related to the subsequent response.

The Board identified ten contributing causes to this accident or resultant response:

1. The preventative and corrective maintenance program did not prevent or correct the buildup of combustible fluids on the salt truck. There is a distinct difference between the way waste-handling and non-waste-handling vehicles are maintained.
2. The fire protection program was less than adequate in regard to flowing down upper-tier requirements relative to vehicle fire suppression system actuation from the Baseline Needs Assessment into implementing procedures. There was also an accumulation of combustible materials in the underground in quantities that exceeded the limits specified in the Fire Hazard Analysis (FHA) and implementing procedures. Additionally, the FHA does not provide a comprehensive analysis that addresses all credible underground fire scenarios including a fire located near the Air Intake Shaft.
3. The training and qualification of the operator was inadequate to ensure proper response to a vehicle fire. He did not initially notify the CMR that there was a fire or describe the fire's location.
4. The CMR Operations response to the fire, including evaluation and protective actions, was less than adequate.
5. Elements of the emergency/preparedness and response program were ineffective.
6. A nuclear versus mine culture exists where there are significant differences in the maintenance of waste-handling versus non-waste-handling equipment.
7. The NWP Contractor Assurance System (CAS) was ineffective in identifying the conditions and maintenance program inadequacies associated with the root cause of this event.
8. The DOE Carlsbad Field Office (CBFO) was ineffective in implementing line management oversight programs and processes that would have identified NWP CAS weaknesses and the conditions associated with the root cause of this event.
9. Repeat deficiencies were identified in DOE and external agencies assessments, e.g., Defense Nuclear Facility Safety Board (DNFSB) emergency management, fire protection, maintenance, CBFO oversight, and work planning and control, but were allowed to remain unresolved for extended periods of time without ensuring effective site response.
10. There are elements of the Conduct of Operations (CONOPS) program that demonstrate a lack of rigor and discipline commensurate with the operation of a Hazard Category 2 Facility.

**Table ES-1: Conclusions and Judgments of Need**

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| **CON 1**: The FSM and Central Monitoring Room Operator (CMRO) did not fully follow the procedures for response to a fire in the underground (U/G). This can be attributed to the complexity of the alarm and communication system, lack of effective drills and training, and additional burdens placed on the FSM due to the lack of a structured Incident Command System (ICS). | **JON 1:** NWP needs to evaluate and correct deficiencies regarding the controls for communicating emergencies to the underground, including the configuration and adequacy of equipment (alarms, strobes, and public address).<br>**JON 2**: NWP needs to evaluate the procedures and capabilities of the FSM and CMRO in managing a broad range of emergency response events through a comprehensive drill and requalification program. |
| **CON 2**: NWP management allows expert-based, rather than a process/systems-based approach to decision making, e.g., shift to filtration during a fire, sheltering decisions, etc. | **JON 3**: NWP needs to evaluate and apply a process/systems-based approach for decision making relative to credible emergencies in the U/G, including formalizing response actions, e.g., decision to change to filtration mode during an ongoing evacuation. |
| **CON 3**: The emergency management program was not structured such that personnel were driven to adequately size up, properly categorize, and classify emergency events.<br>The WIPP (NWP and CBFO) emergency management program is not fully compliant with DOE O 151.1C, *Comprehensive Emergency Management System*, e.g., activation of the EOC, classification and categorization, emergency action levels, implementation of the ICS, training, triennial exercise, etc. Weaknesses in classification, categorization, and emergency action levels (EALs) were previously identified by external reviews and uncorrected. | **JON 4**: NWP and CBFO need to evaluate their corrective action plans for findings and opportunities for improvement identified in previous external reviews, and take action to bring their emergency management program into compliance with requirements.<br>**JON 5**: NWP and CBFO need to correct their activation, notification, classification, and categorization protocols to be in full compliance with DOE O 151.1C and then provide training for all applicable personnel.<br>**JON 6**: NWP and CBFO need to improve the content of site-specific EALs to expand on the information provided in the standard EALs contained in DOE O 151.1C.<br>**JON 7**: NWP and CBFO need to develop and implement an Incident Command System (ICS) for the EOC/CMR that is compliant with DOE O 151.1C and is capable of assuming command and control for all anticipated emergencies. |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| **CON 4**: Actions to be taken by the Operator in the event of a U/G vehicle fire were not clear.<br><br>There were inconsistencies between procedures and training for fire response that led to an ineffective response to the salt haul truck fire. | **JON 8**: NWP needs to review procedures and ensure consistent actions are taken in response to a fire in the U/G.<br><br>**JON 9**: NWP, CBFO and DOE need to clearly define expectations for responding to fires in the U/G, including incipient and beyond incipient stage fires. |
| **CON 5**: NWP and CBFO failed to ensure that training and drills effectively exercised all elements of emergency response to include practical demonstration of competence, e.g., donning of self-rescuers and SCSRs, U/G personnel response to a fire, use of portable fire extinguishers, EOC roles, classification and categorization, notifications and reporting, and allowance of unescorted access for over 500 personnel, etc. | **JON 10**: NWP and CBFO need to develop and implement a training program that includes hands-on training in the use of personal safety equipment, e.g., self-rescuers, SCSRs, portable fire extinguishers, etc.<br><br>**JON 11**: NWP and CBFO need to improve and implement an integrated drill and exercise program that includes all elements of the ICS, including the MRT, First Line Initial Response Team (FLIRT) and mutual aid; unannounced drills and exercises; donning of self-rescuers/SCSRs; and full evacuation of the U/G.<br><br>**JON 12**: NWP needs to evaluate and improve their criteria for granting unescorted access to the U/G such that personnel with unescorted access to the underground are proficient in responding to abnormal events. |
| **CON 6**: NWP preventive and corrective maintenance program did not prevent or correct the buildup of combustible fluids on the salt haul truck. | **JON 13**: NWP management needs to reevaluate and modify the approach to conducting preventative and corrective maintenance on all U/G vehicles such that combustible fluids are effectively managed to prevent the recurrence of fires. |
| **CON 7**: NWP and CBFO management is not adequately considering overall facility impact with regard to operations, emergency response, and maintenance, which affects the safety posture of the facility, e.g., salt haul truck combustible build-up, conversion of the automatic fire suppression system to manual, removal of the automatic fire detection capability, not using fire resistant hydraulic fluid, discontinued use of the vehicle wash | **JON 14**: NWP and CBFO need to develop and implement a rigorous process that effectively evaluates:<br>• changes to facilities, equipment, and operations for their impact on safety, e.g., plant operations review process;<br>• impairment and corresponding compensatory measures on safety-related equipment; and<br>• the impact of different approaches in |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| station, chaining of ventilation doors and an out-of-service regulator and fans, inoperable mine phones, and other non-waste-handling related equipment. | maintaining waste-handling and non-waste-handling equipment.<br><br>**JON 15**: NWP needs to determine the extent of this condition and develop a comprehensive corrective action plan to address identified deficiencies. |
| **CON 8**: NWP and CBFO management have not effectively managed the quantity and duration of out-of-service equipment. | **JON 16**: NWP needs to develop and implement a process that ensures comprehensive and timely impact evaluation and correction of impaired or out-of-service equipment.<br><br>**JON 17**: CBFO needs to ensure that its contractor oversight structure includes elements for comprehensive and timely evaluation and correction of impaired or out-of-service equipment. |
| **CON 9**: NWP management has allowed less than acceptable rigor in the performance of equipment inspections, resulting in the operation of U/G equipment in unacceptable condition. | **JON 18**: NWP needs to develop and reinforce clear expectations regarding the performance of rigorous equipment inspections in accordance with manufacturer recommendations, established technical requirements; corrective action; and trending of deficiencies. |
| **CON 10**: NWP did not ensure the Baseline Needs Assessment (BNA) addressed requirements of DOE O 420.1C and Mine Safety and Health Administration (MSHA) with the results completely incorporated into implementing procedures. | **JON 19**: NWP needs to ensure that all requirements of DOE O 420.1C and MSHA are addressed in the BNA, with the results completely incorporated into implementing procedures and the source requirements referenced, and that training consistent with those procedures is performed. |
| **CON 11**: NWP and CBFO management did not make conservative or risk-informed decisions with respect to developing and implementing the fire protection program.<br><br>There is inadequate fire engineering analysis due to a lack of integration with ventilation design and operations, and U/G operations, for recognizing, controlling, and mitigating U/G fires. | **JON 20**: NWP and CBFO need to perform an integrated analysis of credible U/G fire scenarios and develop corresponding response actions that comply with DOE and MSHA requirements.<br><br>The analysis needs to include formal disposition regarding the installation of an automatic fire suppression system in the mine. |
| **CON 12**: NWP and CBFO have failed to take appropriate action to correct combustible loading issues that were | **JON 21**: NWP and CBFO need to review the combustible control program and complete corrective actions that demonstrate compliance |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| identified in previous internal and external reviews. | with program requirements. These issues remain unresolved from prior internal and external reviews. |
| **CON 13**: NWP and CBFO have allowed housekeeping to degrade and other conditions to persist that potentially impede egress. | **JON 22**: NWP and CBFO need to evaluate and address deficiencies in housekeeping to ensure unobstructed egress and clear visibility of emergency egress strobes, reflectors, SCSR lights, etc. |
| **CON 14**: NWP has not fully developed an integrated contractor assurance system that provides assurance that work is performed compliantly, risks are identified, and control systems are effective and efficient. | **JON 23**: NWP needs to develop and implement a fully integrated contractor assurance system that provides DOE and NWP confidence that work is performed compliantly, risks are identified, and control systems are effective and efficient. |
| **CON 15**: CBFO failed to adequately establish and implement line management oversight programs and processes to meet the requirements of DOE O 226.1B and hold personnel accountable for implementing those programs and processes. | **JON 24**: CBFO needs to establish and implement an effective line management oversight program and processes that meet the requirements of DOE O 226.1B and hold personnel accountable for implementing those programs and processes. |
| **CON 16**: CBFO management does not have adequate communication processes to ensure awareness of issues that warrant attention from all levels of the DOE staff. | **JON 25**: CBFO needs to accelerate the implementation of a mechanism for all levels of CBFO staff to document, communicate, track, and close issues both internally and with NWP.<br><br>**JON 26**: The CBFO Site Manager needs to institutionalize and communicate expectations for the identification, documentation, reporting, and correction of issues. |
| **CON 17**: DOE HQ failed to ensure that CBFO was held accountable for correcting repeatedly identified issues involving fire protection, maintenance, emergency management, work planning and control, and oversight. | **JON 27**: DOE HQ needs to ensure that repeatedly identified issues related to safety management programs (SMPs) are confirmed closed and validated by the local DOE office.<br><br>This process should be considered for application across the DOE complex and include tracking, closure, actions to measure the effectiveness of closure (line management accountability), and trending to identify precursors and lessons learned.<br><br>**JON 28**: DOE HQ should enhance its required oversight to ensure site implementation of the emergency management policy and requirements |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| | are consistent and effective. Emphasis should be placed on ensuring ICSs are functioning properly and integrated exercises are conducted where personnel are evacuated. |
| **CON 18**: DOE HQ failed to ensure CBFO was provided with qualified technical resources to oversee operation of a Hazard Category 2 Facility in a mine. | **JON 29**: DOE HQ needs to develop and implement a process for ensuring that technical expertise is available to provide support in the unique area of ground control, underground construction, and mine safety and equipment.<br><br>**JON 30**: DOE HQ needs to assist CBFO with leveraging expertise from MSHA, in accordance with the DOE/MSHA Memorandum of Understanding (MOU), in areas of ground control, underground construction, and mine safety where DOE does not have the expertise.<br><br>**JON 31**: DOE HQ needs to re-evaluate resources (i.e., funding, staffing, infrastructure, etc.) applied to the WIPP project to ensure safe operations of a Hazard Category 2 Facility. |
| **CON 19**: The Office of Environmental Management Consolidated Business Center (EMCBC) and CBFO failed to ensure support services as described in the Service Level Agreement were provided. | **JON 32**: EMCBC and CBFO need to develop and implement clear expectations and a schedule for EMCBC to provide support in the areas of regulatory compliance, safety management systems, preparation of program procedures and plans, quality assurance, lessons learned, contractor assurance, technical support, DOE oversight assistance, etc. |
| **CON 20**: There are elements of the CONOPS program that demonstrate a lack of rigor and discipline commensurate with operation of a Hazard Category 2 Facility. | **JON 33**: NWP and CBFO need to evaluate and correct weaknesses in the CONOPS program and its implementation, particularly with regard to flow-down of requirements from upper-tier documents, procedure content and compliance, and expert-based decision making. |
| **CON 21**: NWP and CBFO did not analyze and disposition differences between waste-handling and non-waste-handling vehicles for similar hazards and impacts, e.g., allowing a truck in this condition to be at the waste face. | **JON 34**: NWP and CBFO need to identify and control the risk imposed by non-waste-handling equipment, e.g., combustible buildup, manual vs. automatic fire suppression system, fire-resistant hydraulic oil, etc., or treat waste-handling equipment and non-waste-handling equipment the same. |

| | |
|---|---|
| **CON 22**: NWP and CBFO management allowed a culture to exist where there are differences in the way waste-handling equipment and non-waste-handling equipment are maintained and operated. | **JON 35**: NWP and CBFO management need to examine and correct the culture that exists regarding the maintenance and operation of non-waste-handling equipment. |
| **Positive Statement**: All supervisors and employees in the U/G actively used the mine phone to alert other workers of the fire and the need to evacuate before the evacuation alarm was sounded.<br><br>**Positive Statement**: Workers assisted other workers during the evacuation, including helping them to don self-rescuers and SCSRs.<br><br>**Positive Statement**: Personnel in the U/G exhibited detailed knowledge of the underground and ventilation splits.<br><br>**Positive Statement**: NWP on-site medical response was effective in treating personnel. | |

# 1.0 Introduction

## 1.1 Appointment of the Board

On February 7, 2014, an Accident Investigation Board (the Board) was appointed by Matthew Moury, Deputy Assistant Secretary, Safety, Security, and Quality Programs, U.S. Department of Energy (DOE), Office of Environmental Management (EM), to investigate the fire on the EIMCO 985-T15 salt haul truck in the underground at the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico, that occurred February 5, 2014. The Board's responsibilities have been completed with respect to this investigation. The analysis and the identification of the contributing causes, the root cause and the Judgments of Need resulting from this investigation were performed in accordance with DOE Order (O) 225.1B, *Accident Investigations*.

This accident meets Accident Investigation Criteria 2.d.1 of DOE O 225.1B, Appendix A (i.e., estimated loss of or damage to DOE property, including aircraft, equal to or greater than $2.5 million or requiring estimated costs equal to or greater than $2.5 million for cleaning, decontaminating, renovating, replacing, or rehabilitating property).

DOE appointed an Accident Investigation team on February 7, 2014. The accident scene was preserved to the extent practical considering the entries needed to facilitate preparation of the mine for occupancy.

The Board began the investigation on February 10, 2014, completed the investigation on March 8, 2014, and submitted findings to the appointing official on March 11, 2014. The Board concluded that this accident was preventable.

On February 5, 2014, three entries into the underground were performed by the Mine Rescue Team (MRT) to extinguish and overhaul the fire. Nuclear Waste Partnership LLC (NWP) had a procedure for event reporting and investigation, WP 15-MD3102, Rev. 2, *Event Investigation Management Control Procedure*. A written Notification report, EM-CBFO-NWP-WIPP-2014-0001, *Underground Salt Haul Truck Fire*, was transmitted February 7, 2014. NWP took action to establish control of the accident scene by placing security seals on entrances to the above-ground waste-handling area and the mine itself. Subsequent entries were required to be performed by NWP to facilitate the Board's entry on February 13, 2014.

## 1.2 Carlsbad Field Office

The DOE created the Carlsbad Area Office in Carlsbad, New Mexico, in late 1993 to lead the nation's transuranic (TRU) waste disposal efforts. In September 2000, the office was elevated in status to become the Carlsbad Field Office (CBFO). As a field office, CBFO has continued its primary mission of operating WIPP in conformance with the WIPP Land Withdrawal Act (Public Law 102-579 as amended by Public Law 104-201). CBFO is responsible for oversight of the management and operations (M&O) contract for the WIPP site and the National TRU Program. CBFO has taken on additional roles to support the DOE-EM, such as serving as an international center for the study of waste management and enabling the unique capabilities of WIPP to be utilized to support basic scientific research. This includes the Enriched Xenon

Observatory (EXO) laboratory in the north end of the repository. In addition to operations in southeastern New Mexico, the CBFO coordinates the TRU waste characterization and shipping programs at waste-generating sites and national laboratories around the nation.

The organizational components of the CBFO include the Office of the Manager, and the Offices of Site Operations, the National TRU Program, Environment, Safety and Health, Business, Quality Assurance, and Science and International Programs.

## 1.3 Nuclear Waste Partnership LLC

NWP is the M&O for the WIPP facility and the National TRU Program. DOE awarded the contract to NWP on April 20, 2012. NWP is a partnership between URS Energy and Construction, Inc. (URS), the Babcock and Wilcox Company (B&W), and Areva, Inc. (Areva). NWP assumed responsibility for management and operation of the WIPP facility October 1, 2012, after a 90-day transition period. The prior M&O was Washington TRU Solutions, LLC (WTS). WTS and its predecessor entities held the contract from 2000 until NWP took over WIPP operations. WTS was an entity comprised of URS and Weston Solutions, Inc.

Upon transition from WTS to NWP, the management of the WIPP facility did not see a substantial change in management personnel. A new site operations manager from B&W was brought in from the Pantex facility. Additionally, a new business manager was brought in from the B&W Oak Ridge operations. NWP also made revisions to the organizational reporting structure.

## 1.4 Facility Description

DOE was authorized by Public Law 96-164, Department of Energy National Nuclear Security and Military Applications of Nuclear Energy Authorization Act of 1980, to provide a research and development facility for demonstrating the safe, permanent disposal of TRU wastes from national defense activities and programs of the United States exempted from regulations by the U.S. Nuclear Regulatory Commission.

The WIPP Land Withdrawal Act, Public Law 102-579 (as amended by Public Law 104-201), authorized the disposal of 6.2 million cubic feet of defense TRU waste at the WIPP facility. The WIPP facility operates in several regulatory regimes. DOE has authority over the general operation of the facility, including radiological operations prior to closure. The U.S. Environmental Protection Agency (EPA), through its regulations at 40 CFR Parts 191 and 194, certifies the long-term radiological performance of the repository over a 10,000-year compliance period after closure of the facility. The State of New Mexico, through EPA delegation of the Resource Conservation and Recovery Act (RCRA), has issued a Hazardous Waste Facility Permit for the disposal of the hazardous waste component of the TRU waste. Additionally, the Mine Safety and Health Administration (MSHA) is required to perform four inspections per year of WIPP.

WIPP, located in southeastern New Mexico near Carlsbad, was constructed to determine the efficacy of an underground repository for disposal of TRU waste (Figure 1). Disposal operations began in 1999 and are scheduled to continue for 35 years.

**Figure 1: Waste Isolation Pilot Plant near Carlsbad, New Mexico**

## 1.5 Waste Isolation Pilot Plant

The WIPP facility is a geologic repository mined within a bedded salt formation. The underground is 2,150 feet beneath the ground surface. TRU mixed waste management activities underground are confined to the southern portion of the 120-acre mined area.

Four shafts connect the underground area with the surface. The Waste Shaft headframe and hoist are located within the Waste Handling Building and are used to transport TRU mixed waste, equipment, and materials to the repository. The waste hoist can also be used to transport personnel and materials. The Air Intake Shaft and the Salt Handling Shaft provide ventilation to all areas of the mine except for the Waste Shaft station. This area is ventilated by the Waste Shaft itself. The Salt Handling Shaft is also used to hoist mined salt to the surface and serves as the principal personnel transport shaft. The Exhaust Shaft serves as a common exhaust air duct for all areas of the mine (Figure 2).

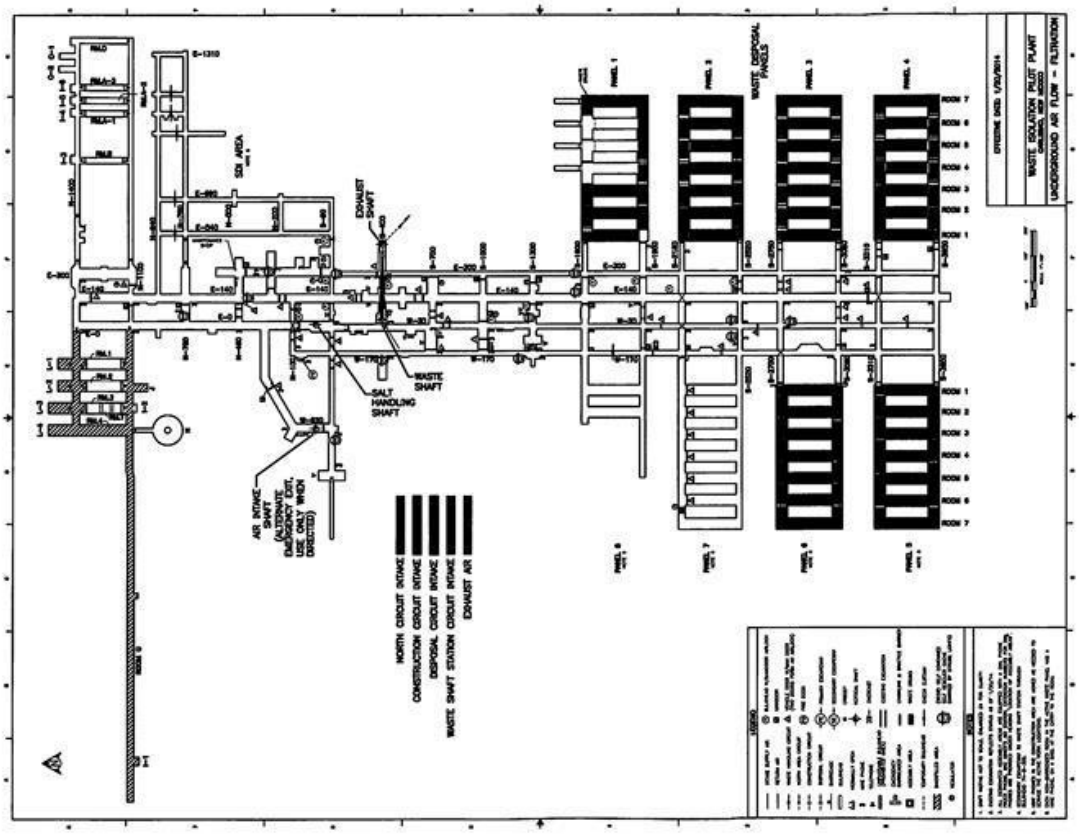



**Figure 2: Mine Layout**

The WIPP underground consists of the waste disposal area, construction area, north area, and Waste Shaft station area. The location of the accident is shown in Figure 3.

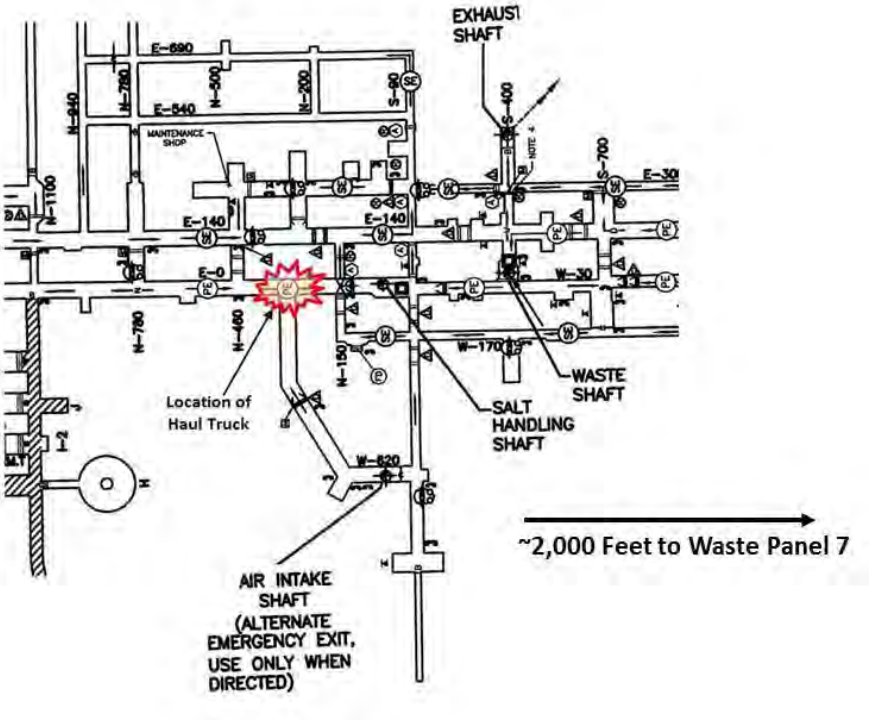



**Figure 3: Location of the Haul Truck during the Fire**

The principle contact-handled (CH) waste operations at the WIPP involve (1) the receipt and disposal of TRU waste, and (2) the mining of underground rooms in which the waste is disposed. In the underground, the waste containers are removed from the waste hoist conveyance, placed on the underground transporter, and moved to a disposal room. In the disposal rooms, the CH waste containers are removed from the transporter and placed in the waste stack. Remote-handled (RH) waste is placed in boreholes in the walls (ribs) of the disposal rooms.

The site has 55 permanent buildings and four temporary buildings (trailers) in operation, one temporary building (lab trailer) in excess status, and various connexes (used for storage). The site buildings provide a total of 358,647 square feet of office and industrial space. Additional leased office space, the Skeen-Whitlock Building, is located in Carlsbad. Approximately 800 workers are assigned to the WIPP, representing the CBFO, the management and operating contractor, the security subcontractor, the warehouse, the document services subcontractor, the information technologies subcontractor, the CBFO Technical Assistance Contractor, Los Alamos National Laboratory-Carlsbad, Sandia National Laboratories-Carlsbad, and the New Mexico Environment Department-Carlsbad. Prominent features of the WIPP site include:

- **Air Intake Shaft**. The primary source of intake for the underground ventilation and also used for emergency egress.
- **Waste Handling Building**. This structure provides a confinement barrier. Ventilation is operated to maintain a negative pressure with high-efficiency particulate air (HEPA) filtration.

- **Waste Hoist**. The Waste Hoist transports waste, material and personnel from the surface to the underground and is designed to prevent an uncontrolled fall or descent of the waste conveyance into the Waste Shaft.
- **Salt Handling Shaft Hoist**. This hoist transports mined salt to the surface, material, and personnel between the surface and the underground.
- **Radiation Monitoring**. Consists of continuous air monitors, fixed air samplers, and other external radiation monitors.
- **Central Monitoring Room**. Provides a monitoring function and must be staffed and operational, with the ability to shift underground ventilation to filtration.
- **Waste Handling Equipment**. Selected items are designated safety class or safety significant.
- **Emergency Services Bay**. Houses the ambulance, rescue truck, and fire engine.
- **Guard and Security Building**. Houses the security monitoring and alarm systems.
- **Parking Lot**. The east portion of the front parking lot is used for employee parking, and the two west rows of the lot are designated for trailer storage and staging of empty transuranic package transporters (TRUPACTs) for DOE carrier transport to the generator sites and trailer maintenance facility.

## 1.6 Background

On February 5, 2014, an underground (U/G) fire involving an EIMCO haul truck 985-T15 (salt haul truck), property ID 74-U-006B, occurred at the DOE WIPP site near Carlsbad, New Mexico. The fire necessitated the evacuation of 86 workers from the U/G, and 13 of the workers required treatment for smoke inhalation, six at the Carlsbad Medical Center (CMC) and seven on-site.

EIMCO Model 985T-15 haul truck 74-U-006B was purchased in May 1985 and has been used continuously over the past 29 years to transport mined salt to the salt hoist for removal from the underground. A second Model 985-15 salt haul truck 74-U-006A is also in operation in the mine. Figure 4 is a photograph of a Model 985 haul truck (74-U-006A)

**Figure 4: EIMCO 985T 15 Haul Truck (74-U-006A)**

The truck has a capacity of 15 tons and is powered by a Deutz V-8 air cooled diesel engine. The truck is equipped with a remote-mounted three-speed (in both forward and reverse) Clark powershift transmission, engine-mounted torque converter, and four-wheel drive with planetary gear wheel ends and integral liquid-cooled brakes. Two 12 volt DC batteries provide electrical power to the vehicle.

The truck when purchased did not include a fire suppression system. The site contractor had an automatic fire suppression system installed sometime before 1995. Due to numerous inadvertent activations, including some which occurred while the vehicle was parked and not running, the site contractor had Southwest Fire Safety switch the automatic system to manual activation in 2003.

In September 2005, there was a fire on this salt haul truck caused by an electrical short, which was extinguished by manually activating the fire suppression system.

The truck contains combustible fluids, including diesel fuel (33-gallon tank capacity); engine oil (3.3 gallons); torque converter/transmission fluid (10.5 gallons); hydraulic fluid for steering, brakes, and the dump box (35 gallons); differential oil (6.25 gallons); wheel end lubricant (2 gallons); and joint lubricant. In the past, trucks were periodically cleaned underground in a wash station but this was taken out of service prior to 2004 because of the difficulty in removing the wash water to the surface.

The truck undergoes a quarterly emissions test, 100-hour preventative maintenance, and 500-hour preventative maintenance. Record review and interviews indicate that the engine has been rebuilt once since it was put into service at the WIPP.

The work history over the last three years includes the above preventative maintenance, a battery replacement, hydraulic hose replacement, and troubleshooting for electrical shorts.

## 1.7 Scope, Purpose and Methodology of the Accident Investigation

The Accident Investigation Board began its activities on February 10, 2014, and completed its investigation on March 8, 2014. The scope of the Board's investigation was to identify relevant facts; analyze the facts to determine the direct, contributing, and root causes of the event; develop conclusions; and determine Judgments of Need for actions that, when implemented, should prevent recurrence of the accident. The investigation was performed in accordance with DOE Order 225.1B, *Accident Investigations*, using the following methodology:

- Facts relevant to the event were gathered through interviews and reviews of documents and other evidence, including photographs and visits to the event scene.
- Facts were analyzed to identify the causal factors using event and causal factors analysis, barrier analysis, change analysis, root cause analysis, and Integrated Safety Management analysis.
- Judgments of Need for corrective actions to prevent recurrence were developed to address the causal factors of the event.

Figure 5 defines the accident investigation terminology used throughout this report.

| Accident Investigation Terminology |
| --- |
| A **causal factor** is an event or condition in the accident sequence that contributes to the unwanted result. There are three types of causal factors: direct cause(s), which is the immediate event(s) or condition(s) that caused the accident; root causes(s), which is the causal factor that, if corrected, would prevent recurrence of the accident; and the contributing causal factors, which are the causal factors that collectively with the other causes increase the likelihood of an accident, but which did not cause the accident.<br><br>The **direct cause** of an accident is the immediate event(s) or condition(s) that caused the accident.<br><br>**Root causes** are the causal factors that, if corrected, would prevent recurrence of the same or similar accidents. Root causes may be derived from or encompass several contributing causes. They are higher-order, fundamental causal factors that address classes of deficiencies, rather than single problems or faults.<br><br>**Contributing causes** are events or conditions that collectively with other causes increased the likelihood of an accident but that individually did not cause the accident. Contributing causes may be longstanding conditions or a series of prior events that, alone, were not sufficient to cause the accident, but were necessary for it to occur. Contributing causes are the events and conditions that "set the stage" for the event and, if allowed to persist or recur, increase the probability of future events or accidents.<br><br>**Event and causal factors analysis** includes charting, which depicts the logical sequence of events and conditions (causal factors that allowed the accident to occur), and the use of deductive reasoning to determine the events or conditions that contributed to the accident.<br><br>**Barrier analysis** reviews the hazards, the targets (people or objects) of the hazards, and the controls or barriers that management systems put in place to separate the hazards from the targets. Barriers may be physical or administrative.<br><br>**Change analysis** is a systematic approach that examines planned or unplanned changes in a system that caused the undesirable results related to the accident.<br><br>**Error precursor analysis** identifies the specific error precursors that were in existence at the time of or prior to the accident. Error precursors are unfavorable factors or conditions embedded in the job environment that increase the chances of error during the performance of a specific task by a particular individual, or group of individuals. Error precursors create an error-likely situation that typically exists when the demands of the task exceed the capabilities of the individual or when work conditions aggravate the limitations of human nature. |

**Figure 5: Accident Investigation Terminology**

# 2.0 The Accident

## 2.1 Description of Work Activity

The WIPP facility is designed for the excavation of eight panels branching off of the main drifts. WIPP uses the concept of "just-in-time excavation" (Figure 6). Just-in-time excavation is based on the concept that when additional room is needed for waste disposal, a new panel would be excavated and ready for use "just in time." This means that each panel would be excavated, filled, and closed in a time frame that would minimize the potential for developing hazardous ground conditions.

Excavation of a new panel is performed by a mining machine that uses a rotary bit to remove the salt. Salt from mining must be removed from the underground and salt haul trucks (trucks) are used to move the salt to the loading pocket where it is dumped and then taken to the surface via the salt hoist.

Panel 7 was completed and certified in late 2013 and CH and RH waste were being disposed in Panel 7 during January and early February 2014.

Panel 8 excavations began after completion of Panel 7 in 2013, and two rooms had been excavated in Panel 8.

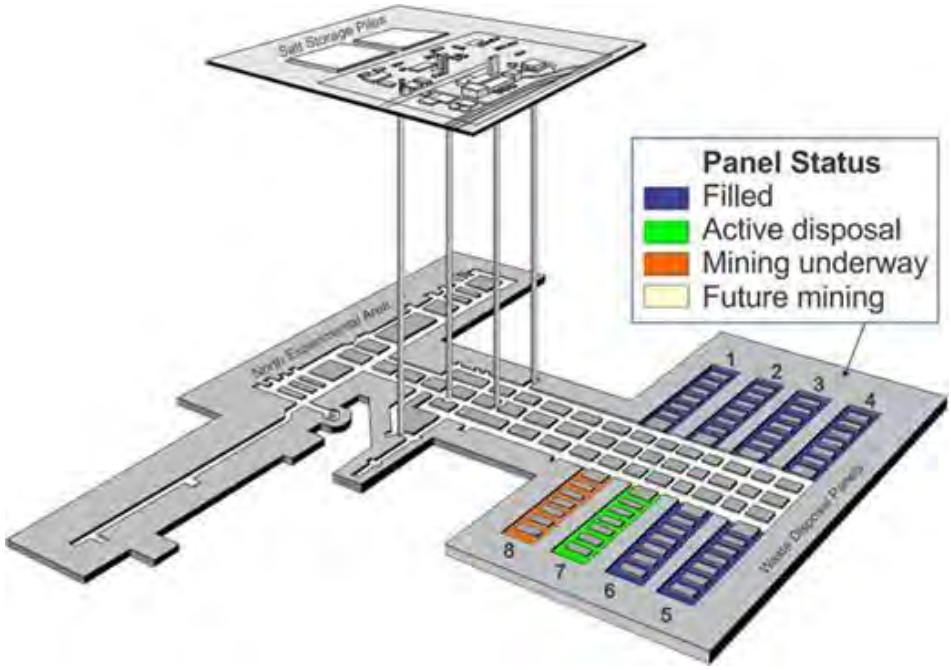



**Figure 6: Panel Layout**

## 2.2 Accident Description

An Operator picked up a load of salt using haul truck 74-U-006B at Panel 8 at approximately 1045 and headed north on W-170 toward the loading pocket (Figure 7) to dump the load. Figure 8 shows the Operator's route from Panel 8 to the scene of the fire. He turned right on S-90, left on E-0, dropped half of his load at the loading pocket, continued north in E-0, and passed N-150 to drop the rest of his load. The Operator pulled into N-300, backed up into E-0, and unloaded the rest of the truck. As the Operator lowered the bed, he looked back to see if it was clear of muck. It was at this point that he noticed an orange glow and then flames between the engine and the dump sections of the truck.

**Figure 7: The Loading Pocket "The Grizzly"**

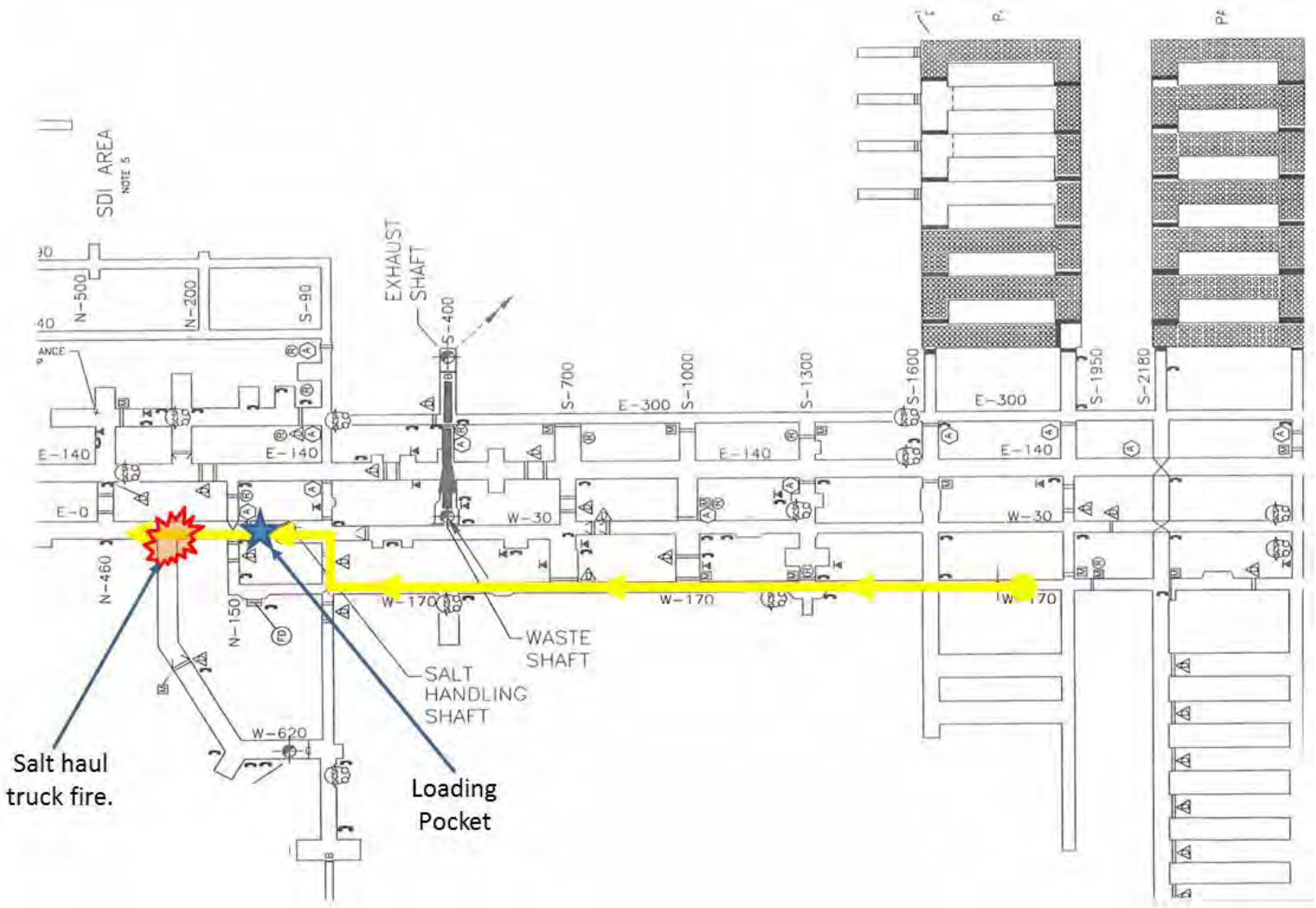



**Figure 8: Route of Haul Truck from Panel 8 to Accident Scene**

## 2.3 Event Chronology

**Table 1: Chronology of the Salt Haul Fire Events**

| Date and Time (hours) (MST) | EVENT |
|---|---|
| May 1, 1985 | EIMCO Salt Haul Truck 74-U-006B (Truck 6B) is purchased. Does not include an onboard fire suppression system (FSS). |
| 1985 – 2013 | Truck 6B is in service, receives scheduled maintenance, refurbished at least once (July 2004). |
| ~1993 | Automatic FSS is added to Truck 6B. |
| October 2003 | Due to inadvertent actuations, the automatic FSS is converted to manual activation. |
| ~2003 - 2004 | Wash station taken out of service/replacement wash station not in service. |
| 2004 | Truck 6B engine is rebuilt. |
| October 23, 2013 | Quarterly emissions tests were performed on Truck 6B, results satisfactory. |
| November 26, 2013 | 100 hour preventative maintenance is performed on Truck 6B, results satisfactory. |
| December 17, 2013 | Batteries are replaced on Truck 6B via an expedited work package. |
| December 21, 2013 | 500 hour preventative maintenance is performed on Truck 6B, results satisfactory. |
| January 20, 2014 | 100 hour preventative maintenance is performed on Truck 6B, results satisfactory. |
| January 21, 2014 | Replaced hydraulic hose on Truck 6B per expedited work package. |
| January 24, 2014 | Quarterly emissions tests were performed on Truck 6B, results satisfactory. |
| January 24 – February 5, 2014 | Truck 6B was in service, transporting salt from the mined panels to the loading pocket for dumping and then removal from the underground via the salt hoist. |

| Date and Time (hours) (MST) | EVENT |
|---|---|
| February 5, 2014 0000 hours | The Facility Shift Manager (FSM) directed the Central Monitoring Room Operator (CMRO) to put ventilation system in maintenance bypass mode, filtration is enabled. |
| February 5, 2014 0555 hours | Salt hoist checks are completed. |
| February 5, 2014 0816 | Contact –handled (CH) waste bay was configured for waste-handling (WH) mode. |
| February 5, 2014 0834 | CMRO disabled filtration for underground (U/G) local processing unit (LPU) testing. |
| February 5, 2014 0835 | LPU testing was unsatisfactory (results required a manual shift to filtration upon a loss of power scenario). |
| February 5, 2014 0848 | FSM directed CMRO to configure the shaft access area (SAA) and U/G for CH waste handling. |
| February 5, 2014 0859 | FSM directed CMRO to configure the SAA and U/G for remote handled (RH) waste handling. |
| February 5, 2014 1007 | FSM directed CMRO to configure the RH bay for waste handling. |
| February 5, 2014 ~1045 | Salt Haul Truck Operator (Operator) was at Panel 8 in Truck 6B and was loaded with salt (last load before lunch). |
| February 5, 2014 ~1046 | Operator headed down W-170 in Truck 6B towards the loading pocket. |
| February 5, 2014 ~1046 | Operator turned right on S-90 through the bulkhead, turned left into E-0, and dropped his load of salt at the loading pocket. Not all discharged into the loading pocket. |
| February 5, 2014 ~1047 | Operator headed up north on E-0 and passed N-150 to drop rest of the load on the floor of the drift. |
| February 5, 2014 ~1048 | Operator pulled Truck 6B into N-300, backed up to the rib, and raised the bed on the dump portion of Truck 6B to unload the rest of the load. |
| February 5, 2014 ~1048 | As the Operator was lowering bed, he looked to see if all the muck (salt) was clear of the bed. |

| Date and Time (hours) (MST) | EVENT |
|---|---|
| February 5, 2014 ~1048 | Operator noticed an orange light and then flames coming from the bottom of the truck in the area between the tractor and the dump. |
| February 5, 2014 ~1048 | Operator stopped truck, put on the brake, and shut off the engine. |
| February 5, 2014 ~1049 | Operator got off the truck and grabbed the truck's portable fire extinguisher. |
| February 5, 2014 ~1050 | Operator walked around the truck and discharged the portable fire extinguisher into a hole in the area where he had observed the flames. He also discharged it underneath the truck. |
| February 5, 2014 ~1050 | The fire was not extinguished, so Operator dropped the portable fire extinguisher and activated the onboard FSS on the truck. |
| February 5, 2014 ~1050 | Operator was unsure if the FSS actuated, observed a large puff of smoke (or suppressant). |
| February 5, 2014 ~1050 | Operator was increasingly alarmed and walked to the nearest mine phone (out of the smoke), called Maintenance and then his Supervisor to inform them of the fire. |
| February 5, 2014 ~1050 | Two U/G Services workers begin to respond from their office at S-550/W-30 and the Supervisor responded from S-3080/W-30. |
| February 5, 2014 ~1050 | An U/G Services worker in the office called the CMRO and told the CMRO that there was a fire at N-150/E-0, that they were getting smoke in the office, and to let everyone in the U/G know to get to the waste hoist. |
| February 5, 2014 ~1050 | Operator entered the airlock bulkhead at N-150. |
| February 5, 2014 ~1050 | Two U/G Services personnel attempted to push a 300-pound wheeled fire extinguisher to the airlock at E-0/N-150; as they began to open the airlock, their carbon monoxide monitor alarmed and they saw smoke begin to "boil in" under the outer airlock. |
| February 5, 2014 ~1050 | U/G Services personnel arrived in the area of the fire (brought a carbon monoxide monitor and their self-rescuers). |

| Date and Time (hours) (MST) | EVENT |
|---|---|
| February 5, 2014 ~1051 | CMRO sounded the emergency evacuation alarm (yelp) for approximately two seconds, stated that there was a fire (no location), and that personnel should evacuate via the waste hoist. The alarm and instruction could not be heard and/or understood throughout the U/G. The CMRO operator forgot to activate the emergency evacuation strobe lights. |
| February 5, 2014 ~1052 | Supervisor, Operator, and two U/G services workers decided that the carbon monoxide level was too high to fight the fire and decided to evacuate via W-170, S-1950, W-30, S-1000, and E-140, but encountered thick smoke. They encountered others enroute and informed them of the need to evacuate and to don their self-rescuers. |
| February 5, 2014 ~1058 | CMRO was directed by FSM to change ventilation to filtration mode, believing this would reduce both the fire and smoke. This caused significant changes in air flow and smoke in the U/G. |
| February 5, 2014 ~1051 - 1134 | Workers throughout the U/G were attempting to evacuate the mine in response to the alarm and announcement, what they heard over the mine phones, and/or interactions with other personnel.<br><br>Some workers encountered difficulties (heavy smoke, strobes not on or not working, smoke in areas expected to have "good" air, obscured evacuation reflectors) and improvised routes to the waste hoist, at times cutting holes in ventilation curtains.<br><br>Workers reported near-collisions between personnel, carts, and other equipment.<br><br>Not all workers donned self-rescuers at the first indication of fire (it appears that three never donned them at all) and some had difficulty opening and/or donning self-rescuers or self-contained self-rescuers (SCSRs).<br><br>Workers helped each other don and check self-rescuers and SCSR and made their way in the heavy smoke to the waste hoist. |
| February 5, 2014 ~1103 | The FSM activated the Emergency Operations Center (EOC). CBFO Facility Representative (FR) notified by CBFO Security Manager; the FSM notified CBFO FR at 1135. The EOC did not classify or categorize the event as an operational emergency, and did not notify the DOE HQ watch office. |
| February 5, 2014 ~1108 | EOC held briefing on the fire location and status. |

| Date and Time (hours) (MST) | EVENT |
|---|---|
| February 5, 2014 ~1101 | The first evacuation of workers via the waste hoist (mantrip) to the surface was underway. |
| February 5, 2014 ~1111-1112 | Mine Safety and Health Administration (MSHA) and the State Mine Inspector were notified of the event. |
| February 5, 2014 ~1115 | The CMRO suspended surface waste handling activities. |
| February 5, 2014 ~1120 | The CMRO activated the Mine Rescue Team (MRT). |
| February 5, 2014 ~1125 | The second mantrip was made at the waste hoist. |
| February 5, 2014 ~1126 | The Joint Information Center (JIC) was activated. |
| February 5, 2014 ~1130 | Mine rescue team made a request to the Intrepid and Mosaic (local potash mining companies) to put their MRTs on standby for support. |
| February 5, 2014 ~1134 | The third and final mantrip was made at the waste hoist. |
| February 5, 2014 ~1134 | Full accountability of the U/G was achieved. |
| February 5, 2014 ~1144 | One ambulance and two Emergency Safety Technicians (ESTs) were on scene. FSM contacted Carlsbad Fire Department (CFD) for additional transportation support. |
| February 5, 2014 ~1151 - 1251 | Six workers were examined by site medical personnel and were transferred via ambulance to the Carlsbad Medical Center (CMC) for observation (possible smoke inhalation). |
| February 5, 2014 ~1147 | CMRO secured U/G ventilation. |
| February 5, 2014 ~1125 | Seven additional workers were examined by the site nurse but additional medical attention was not needed. |
| February 5, 2014 ~1311 | CMRO halted release of waste shipments to WIPP. |

| Date and Time (hours) (MST) | EVENT |
|---|---|
| February 5, 2014 ~1312 | CMRO shifted ventilation to CH HVAC to "once through" ventilation (versus recirculation) due to smoke upcasting in the waste hoist shaft and into the CH bay. |
| February 5, 2014 ~1336 | MSHA arrived onsite to support DOE in accordance to the Memorandum of Understanding (MOU). |
| February 5, 2014 ~1348 | MSHA issued a K-Order to WIPP to obtain the approval of the MSHA representative regarding any plan to recover the mine. |
| February 5, 2014 ~1420 | All workers were released from the CMC. |
| February 5, 2014 ~1420 | The CMRO continued monitoring air quality at the mine shafts. |
| February 5, 2014 ~1614 | CMRO shifted from waste handling mode for Technical Safety Requirements (TSR) compliance. |
| February 5, 2014 ~1722 | The first MRT (MRT1) entered the U/G via the air intake shaft. |
| February 5, 2014 ~1746 | MRT1 reported gas checks at the station level (0 percent methane, 0 percent carbon monoxide, oxygen 21 percent). |
| February 5, 2014 ~1825 - ~1900 | MRT1 arrived at the haul truck. No fire was detected but embers were noticed on the front tires, and ground checks were performed. Discharged portable fire extinguishers on the embers. |
| February 5, 2014 ~1958 | MRT1 arrived back at surface. |
| February 5, 2014 ~2205 | The second MRT (MRT2) entered the U/G via the air intake shaft. |
| February 5 - 6, 2014 ~2208-~0059 | MRT2 performed air quality checks, checked and/or closed ventilation louvers and doors. |
| February 5, 2014 ~2300 | MRT2 drove U/G rescue truck to the scene, discharged all foam fire suppressant, and noted that the fire appeared to be out. |
| February 6, 2014 ~0059 | MRT2 arrived back at surface and U/G accountability was declared complete. |

| Date and Time (hours) (MST) | EVENT |
|---|---|
| February 6, 2014 ~0105 | Event is terminated, EOC and JIC are deactivated. |
| February 6, 2014 | Initial all-hands meetings hosted by CBFO and NWP management. |
| February 7, 2014 1000 | Critique meeting was held to gather facts and establish the initial timeline. |
| February 7, 2014 | Occurrence Reporting and Processing System (ORPS) notification report filed. |
| February 7, 2014 | Accident Investigation Board appointed. |

# 3.0 Emergency Response

## 3.1 Accident Response

Upon noticing the fire, the Operator stopped the truck, shut off the engine, set the brake, and exited the vehicle, taking a portable fire extinguisher which was mounted on the left front fender. The Operator proceeded to the opposite side of the vehicle, near the articulation joint and attempted to extinguish the fire by discharging the fire extinguisher into the area where the Operator had observed the fire (Figure 9).

**Figure 9: Photo Showing the Area on the Salt Haul Truck where the Fire Extinguisher was Discharged**

When this proved unsuccessful, the Operator attempted to actuate the onboard manual fire suppression system, which resulted in a large puff of either smoke or suppressant. This also proved ineffective. At this point, the Operator proceeded to the nearest mine phone (out of smoke) and called Maintenance to report the fire. At approximately 1050, the Operator entered the bulkhead N-150 airlock and encountered two U/G Services workers who had come from the S-550/W-30 office to assist. They had become aware of the fire via the Operator's conversation over the mine phone, which could be heard throughout the U/G, and had observed smoke in W-30 coming south from the Salt Handling Shaft area (Figure 10). Another member of U/G Services called the CMRO to report the fire and indicated that an evacuation was necessary.

At 1051, the CMRO sounded the evacuation "yelp" alarm for approximately two seconds, and then made a public address (PA) system announcement that there was a fire in the underground and for all workers to evacuate via their area egress points. A subsequent announcement directed

personnel to the waste hoist. The CMRO forgot to activate the emergency egress lights until he received a call from the bottom lander, which contributed to U/G personnel delays in exiting.

At 1058, the Facility Shift Manager (FSM) directed the CMRO to change ventilation to filtration mode believing this would reduce both the fire and smoke. This caused changes in air flow and smoke in the U/G and contributed to confusion as people attempted to make their way to the waste hoist. Workers throughout the U/G attempted to evacuate the mine in response to the alarm and announcement, what they heard over the mine phones, and/or interactions with other personnel.

**Figure 10: Smoke Visible Exiting through the Salt Shaft**

At 1103, the FSM activated the Emergency Operations Center (EOC) and notified the CBFO Facility Representative. The EOC did not classify or categorize the event as an operational emergency, and did not notify the DOE-HQ watch office.

At 1108, the EOC held a briefing on the fire location and status and the first evacuation of workers via the waste hoist (mantrip) to the surface was underway. MSHA and the State Mine Inspector were notified of the event at 1112.

At 1115, the CMRO suspended surface waste-handling activities and the CMRO activated the Mine Rescue Team at 1120.

Workers continued to be evacuated from the U/G, with the second mantrip at the waste hoist at 1125 and the third and final mantrip at 1134. Full accountability of all personnel was achieved at 1134.

**Figure 11: 300-Pound Extinguisher in the Underground**

The Joint Information Center (JIC) was activated at 1126 and all external notifications were completed. As noted above, because the site did not classify and categorize the event as an operational emergency, the DOE HQ watch office was not notified.

The two U/G Services workers attempted to push a 300-pound wheeled fire extinguisher (see Figure 11) to the airlock at E-0/N-150. When the workers opened the airlock, their ITX (carbon monoxide) monitor alarmed and the smoke worsened. The Operator's supervisor (after notifying his room closure crew and the Mine Manager of the fire)

arrived at the scene via W-170. The group realized at this point that the fire was beyond their control. They then began moving south in W-170 towards the waste hoist, at one point having to cut a ventilation curtain to continue toward the waste hoist. During the workers' egress they encountered other personnel in carts. They informed them of the fire and to put on their self-rescuers. The group then crossed into E-140 and travelled to the Waste Shaft where they were evacuated to the surface.

During the evacuation, some personnel encountered difficulties (heavy smoke, strobes not on or not working, smoke in areas expected to have "good" air, obscured evacuation reflectors) and had to improvise routes to the waste hoist, at times cutting holes in ventilation curtains.

There were a number of near-collisions between personnel, carts, and other equipment reported. Additionally, not all personnel donned their self-rescuers at the first indication of fire (three reported that they never donned them at all) and others had difficulty opening and/or donning self-rescuers or SCSRs.

There were several reports that personnel helped each other don and check self-rescuers and SCSRs and make their way in the heavy smoke to the waste hoist.

Between 1151 and 1251, six personnel were examined by site medical personnel and transferred via ambulance to the CMC for observation (possible smoke inhalation). Seven additional personnel were examined by the site nurse, but no further treatment was necessary. At 1420, the six workers were released from the CMC.

At 1336, MSHA arrived onsite and issued a K-Order to obtain the approval of the MSHA representative regarding any plan to recover the mine.

The first MRT entered the U/G via the Air Intake Shaft at 1722, conducted gas checks, and upon arrival at the truck found no fire but the presence of embers on the front tires, and performed ground checks. They discharged four portable fire extinguishers on the embers. They arrived back at the surface at 1758.

The second MRT entered the U/G via the Salt Handling Shaft at 2205, performed air quality checks, checked and/or closed ventilation louvers and doors, and drove the U/G rescue truck to the scene where they discharged all foam fire suppressant, and noted that the fire appeared to be out. They arrived back at the surface at 0059, on February 6, 2014.

At 0105, the event was terminated and the EOC and JIC were deactivated.

## 3.2 Emergency Management Program Implementation

The WIPP Emergency Management Program is implemented through WP 12-9 series emergency response procedures, and the WP 12-ER series emergency management procedures. These procedures are designed to provide guidance, define the responsibilities for Operational Emergency (OE) categorization and classification, and define the organization structure and responsibilities. The WP 12-9 series identifies actions to activate the emergency response organizations and respond to emergencies, and defines the lines of authority. Additionally, WP

12-ER3906, *Categorization and Classification of Operational Emergencies*, identifies Emergency Action Levels (EAL) that provides the criteria to categorize an OE.

During on-site emergency conditions, the FSM is in control of the facility, and is the Incident Commander. The FSM is also responsible for event categorization and classification, and activates the EOC. When the EOC is activated, a Crisis Manager assists the FSM with emergency actions. WIPP also has a Central Monitoring Room Operator (CMRO) that is responsible for reporting information concerning events to the FSM and notifying WIPP emergency response teams (ERTs) and support groups.

The Board reviewed execution of the WIPP Emergency Management Program and identified the following facts via witness statements, personnel interviews and program documents.

### 3.2.1 Fire Response and Evacuation

During the event, the evacuation alarm was not activated for a full five seconds and the evacuation strobe lights were not turned on as required by WP 12-ER4911, *Underground Fire Response*. Additionally, the CMRO did not inform personnel of the fire location or to suspend all U/G operations. Interviews with NWP employees also stated that the voice coming from the PA system was garbled and not understandable.

The CMRO was not immediately notified of the fire event because the Operator had first contacted the Maintenance Department, and then notified his supervisor via the mine phone. Underground Services heard of the fire via the mine phone and notified the CMRO. The WIPP Underground Fire Response procedure requires that the emergency notification be made to the CMR first.

The U/G ventilation was shifted to filtration mode. This unannounced shift resulted in an unexpected condition for the U/G personnel as they attempted to evacuate the mine. U/G personnel are familiar with ventilation mode changes, and could tell by movement of louvers and reduction of airflow in evacuation paths that ventilation was changed. Interviews with workers indicated that the change in ventilation mode resulted in an increase in anxiety for the U/G personnel.

Large quantities of material were staged haphazardly throughout the mine. The contents of the maintenance shop lined both sides of the drift. Additionally, the U/G green and red reflectors that provide an indication of where to proceed during an evacuation were not effective. Some were obscured by being placed under the mesh fence along the ribs, while others were hidden from sight by other material stored in the mine. The Board also identified that these reflectors were covered in soot from the fire, and were located at irregular spacing (Figure 12).


**Figure 12: Obscured Reflectors**

**Analysis**

Procedural non-compliances and off-script actions by the CMRO and the FSM represent a response that could have endangered workers as they attempted to evacuate. The unannounced change in ventilation to filtration mode was not in any procedure and quite possibly contributed to higher local concentrations of smoke and carbon monoxide in the drifts. The procedure used in the CMR did not anticipate a full spectrum of potential emergency situations. This requires the FSM to make decisions based on his expert knowledge in a given situation. Communication problems and unclear announcements contributed to confusion throughout the mine. The Board determined that there was a lack of effective drills and training, there was complexity of the alarm and communication system, and there were additional burdens placed on the FSM due to the lack of a structured Incident Command System. The Board also determined that the poor housekeeping observed throughout the mine had a negative impact on the ability of workers to navigate to the egress point in the reduced visibility environment.

**CON 1**: The FSM and Central Monitoring Room Operator (CMRO) did not fully follow the procedures for response to a fire in the U/G. This can be attributed to the complexity of the alarm and communication system, lack of effective drills and training, and additional burdens placed on the FSM due to the lack of a structured Incident Command System (ICS).

**JON 1:** NWP needs to evaluate and correct deficiencies regarding the controls for communicating emergencies to the underground, including the configuration and adequacy of equipment (alarms, strobes, and public address).

**JON 2**: NWP needs to evaluate the procedures and capabilities of the FSM and CMRO in managing a broad range of emergency response events through a comprehensive drill and requalification program.

**CON 2**: NWP management allows expert-based, rather than a process/systems-based approach to decision making, e.g., shift to filtration during a fire, sheltering decisions, etc.

**JON 3**: NWP needs to evaluate and apply a process/systems based approach for decision making relative to credible emergencies in the U/G, including formalizing response actions, e.g., decision to change to filtration mode during an ongoing evacuation.

### 3.2.2 Emergency Categorization and Classification

During the event, the EOC was activated at approximately 10 minutes into the incident. EOC staff is considered the Crisis Management Team (CMT). This team includes a Crisis Manager, Deputy Crisis Manager, Safety Representative, Operations Representative, EOC Coordinator, Consequence Assessment Support, and a DOE representative called the CBFO Emergency Representative (CER). Also, the following support personnel may be located in the EOC: Public Affairs Coordinator, Human Resources Manager, Safety Coordinator, and Security Coordinator.

As stated earlier, during an incident the FSM has full authority and responsibility for coordinating all emergency response measures. The contractor's plans do not allow the FSM to transfer the Emergency Director position to a more senior official such as the Crisis Manager in the EOC. In a previous HS-45 assessment of August 2012, it was recommended that WIPP consider transferring some of the FSM's responsibility to the EOC's Crisis Manager to relieve some of the burden on the FSM. For this event:

- The fire event was not classified as Operational Emergency;
- The fire event was reported into the ORPS as a Significance Category 2, *Any Fire Emergency or Fire Incident in a Nuclear Facility*; and
- The DOE Facility Representative was notified by the FSM approximately 15 minutes after discovery.

**Analysis**

The current response organization does not provide the recommended Incident Command System (ICS) span of control for the FSM position during a large incident and could constrain the FSM in making quick and sound decisions. The Board recommends that WIPP should reevaluate the Emergency Response Organization (ERO) structure and responsibilities

NWP chose not to classify this event as an OE, although WIPP procedure WP 12-ER3906, *Categorization and Classification of Operational Emergencies*, provides criteria for the FSM to do so. Additionally, the Crisis Manager failed to ensure that the event had been categorized correctly. This event represented a facility evacuation in response to an actual occurrence that required time-urgent response by specialist personnel. The WIPP emergency response structure diminished the ability of the FSM to focus on strategic and tactical response. Eighty-six workers were in the U/G and a total of 13 workers were treated; six transported to a local hospital and seven treated on-site. Had an OE been declared, required notification to DOE-HQ could have

possibly been made in a timely manner and would have activated additional DOE assets to be placed on standby to assist if the situation were to deteriorate further.

**CON 3**: The emergency management program is not structured such that personnel are driven to adequately size up, properly categorize, and classify emergency events.

The WIPP (NWP and CBFO) emergency management program is not fully compliant with DOE O 151.1C, *Comprehensive Emergency Management System*, e.g., activation of the EOC, classification and categorization, emergency action levels, implementation of the ICS, training, triennial exercise, etc. Weaknesses in classification, categorization, and emergency action levels (EALs) were previously identified by external reviews and uncorrected.

**JON 4**: NWP and CBFO need to evaluate their corrective action plans for findings and opportunities for improvement identified in previous external reviews, and take action to bring their emergency management program into compliance with requirements.

**JON 5**: NWP and CBFO need to correct their activation, notification, classification, and categorization protocols to be in full compliance with DOE O 151.1C and then provide training for all applicable personnel.

**JON 6**: NWP and CBFO need to improve the content of site-specific EALs to expand on the information provided in the standard EALs contained in DOE O 151.1C.

**JON 7**: NWP and CBFO need to develop and implement an Incident Command System (ICS) for the EOC/CMR that is compliant with DOE O 151.1C and is capable of assuming command and control for all anticipated emergencies.

### 3.2.3 Training, Qualifications, Drills & Exercise

Some U/G workers recognized the need to don self-rescuers at the first indication of a fire; however, many workers were unable to open and don the self-rescuers and SCSRs. One worker stated that he did not want to don the SR. Evacuation drill exercises did not include donning self-rescuers and SCSRs. Evacuation drill exercises included long duration yelps and the use of strobe lights. Fully integrated exercises involving all of WIPP's assets have not been conducted. Some qualified FSMs had not received Incident Command System training, even though they are expected to perform in that capacity during an emergency. Additionally, there is no position-specific training for the various EOC roles and responsibilities. The Facility Operations training week had been discontinued.

The Operator that responded to the fire did not receive hands-on training in the use of a portable fire extinguisher. During qualification, the Operator did receive a signature indicating training in the operation of the onboard manual fire suppression system. However, recent training provided as an updated portion of General Employee Training (GET), as well as the *Underground Fire Response* procedure, stressed the use of a portable extinguisher for incipient fire response.

The Board identified 506 personnel with unescorted access to the mine. Many of these personnel rarely visit the mine and possess only the minimum required training for mine access.

### 3.2.4 Fire Brigade and Fire Department Interface

The Mine Rescue Teams were activated. Both teams entered the mine. The Mine Rescue Teams extinguished smoldering embers from the fire using the foam unit mounted on the U/G Rescue Truck. The FSM maintained incident command of the Fire Brigade, as well as being RCRA Emergency Coordinator throughout the emergency.

### 3.2.5 Facilities and Equipment

Underground workers have handheld fire extinguishers available throughout the mine. A 300-pound wheeled dry chemical fire extinguisher is available in the U/G. The U/G Rescue Truck is equipped with a 300-pound dry chemical extinguisher and a 150- gallon foam extinguisher. During the fire, U/G personnel attempted to drag the 300-pound wheeled extinguisher to the fire until they elected to stop due to an increase in carbon monoxide levels.

**Analysis**

Several deficiencies were identified in training, qualifications, and drills. The Operator had not had hands-on training on the use of a portable fire extinguisher. There is a multitude of fire suppression equipment staged in the underground, but there is no clear fire-fighting strategy developed to inform personnel how to employ it. During evacuation drills and exercises, it was common for the evacuation alarms (yelps) to continue for a long period of time (greater than five minutes). Additionally, the evacuation strobe lights would be on during the entire drill or exercise. The absence of alarms and strobe lights during the fire event contributed to U/G personnel being unsure why they were evacuating and what they should be doing. During evacuation drills, WIPP workers were not required to demonstrate the donning of self-rescuers and SCSRs in the U/G. Evidence from the accident scene revealed many difficulties that employees encountered in attempting to utilize self-rescuers and SCSRs.

The Board was unable to determine the need for granting unescorted mine access to 506 personnel. Additionally, the Board questions if all of the 506 personnel possess the requisite knowledge to respond appropriately in an emergency situation.

**CON 4**: Actions to be taken by the Operator in the event of a U/G vehicle fire were not clear.

There were inconsistencies between procedures and training for fire response that led to an ineffective response to the salt haul truck fire.

**JON 8**: NWP needs to review procedures and ensure consistent actions are taken in response to a fire in the U/G.

**JON 9**: NWP, CBFO and DOE HQ need to clearly define expectations for responding to fires in the U/G, including incipient and beyond incipient stage fires.

**CON 5**: NWP and CBFO failed to ensure that training and drills effectively exercised all elements of emergency response to include practical demonstration of competence, e.g., donning of self-rescuers and SCSRs, U/G personnel response to a fire, use of portable fire extinguishers, EOC roles, classification and categorization, notifications and reporting, allowance of unescorted access for over 500 personnel, etc.

**JON 10**: NWP and CBFO need to develop and implement a training program that includes hands-on training in the use of personal safety equipment, e.g., self-rescuers, SCSRs, portable fire extinguishers, etc.

**JON 11**: NWP and CBFO need to improve and implement an integrated drill and exercise program that includes all elements of the ICS, including the MRT, First Line Initial Response Team (FLIRT) and mutual aid; unannounced drills and exercises; donning of self-rescuers/SCSRs; and full evacuation of the U/G.

**JON 12**: NWP needs to evaluate and improve their criteria for granting unescorted access to the U/G such that personnel with unescorted access to the underground are proficient in responding to abnormal events.

### 3.2.6 Medical Response

One ambulance and two Emergency Service Technicians (ESTs) responded to the top of the Waste Shaft. Thirteen employees were assessed by medical staff. Of those assessed, six employees displayed symptoms of carbon monoxide exposure and were transported to the CMC.

The following medical documentation regarding six NWP employees was made available to the DOE Chief Medical Officer for review:

- WIPP Emergency Medical Services (EMS) Service Reports;
- WIPP Personal and Occupational History Forms;
- Emergency Department Physician Documentation from the CMC;
- Discharge Instructions from the CMC;
- Medical Reconciliation Forms from the CMC to be provided to the next provider of medical services, with emphasis on prescribed medications;
- Individual Encounter Forms from TRU Solutions Health Services;
- Worker's Injury/Illness Visit forms; and
- DOE Health Care Assets, Mutual Aid Agreements, Terrorism Response-Related Expertise.

**Analyses**

The above referenced information was made available for six WIPP workers, although the documentation was incomplete for one of the individuals, in that case consisting only of Discharge Instructions from the CMC. The totality of that information resulted in observations in several areas.

**Processes**

Emergency medical support services appeared to be in place to address mine-related hazards, including fire.

- Staffing of EMS personnel who could potentially be activated for off-site events at times when they would be needed on-site was unclear.
- The use of written protocols by on-site nursing staff and EMS personnel was documented, but indications for communications to/from the Incident Commander, the "on-shift EST Coordinator," were not.
- Measures such as the availability of escape respirators were demonstrated. Limited information was available regarding the distribution of escape respirators or fit-testing to ensure their effectiveness. In particular, the medical documentation provided by WIPP on-site medical personnel and emergency medical technician (EMT)-level services only specified the use of escape respirators in a minority of the six cases treated for inhalational injuries.
- WIPP EMS was limited to Basic Life Support, rather than Advanced Cardiac Life Support, which would generally prevent the responding personnel from intubating workers with significant respiratory injuries or distress.

**Response**

- Information was provided that reflected a coordinated medical response involving on-site medical personnel and EMT-level services for the stabilization and transport of injured personnel to the CMC.
- Limited information was made available regarding the apparent delay between the call being "received" by WIPP EMS (i.e., 1051) and the activation of WIPP EMS (i.e., 1147) nearly an hour later.

**Quality of Care**

- Efforts to assess health effects, treat symptoms of affected workers, and speed their return to work were evident. In particular, medical documentation on-site, during transport, and following arrival at the CMC was comprehensive, addressing occupational exposures and evaluation of both the underlying medical histories of affected employees and the results of laboratory and radiographic tests for inhalational injuries.
- Follow-up medical evaluations by WIPP were noteworthy for their consistency across all affected workers, their aggressive management of symptoms, and their effectiveness in return-to-work of affected workers.
- Information was made available to all site personnel and the individuals directly involved via the Employee Assistance Program (EAP) on February 9, 2014. Subsequent EAP counseling was available to groups and individuals from February 11 through February 13, 2014.

# 4.0 Maintenance Program

Maintenance at WIPP is governed by WP 10-WC3011, *Work Control Process*, Rev. 31, effective October 18, 2013, and WP 10-WC3010, *Preventive Maintenance Controlled Document Processing*. Preventive maintenance is initiated through the Computerized History and Maintenance Management System (CHAMPS), based on required frequency. Work planners, along with a planning team in some cases, further develop the activity level work control document and participate in development of a job hazard analysis.

## 4.1 Salt Haul Truck Maintenance

The EIMCO 985 series manufacturer service manual provides a recommended maintenance regimen, including a note that states: "The time intervals specified in the following maintenance schedule may be shortened, according to the severity of working conditions. These intervals may not, however, be lengthened unless otherwise stated without prior consultation with the EIMCO service representative." The recommended maintenance regimen is as follows:

- Every shift or every 10 hours of operation, prior to operation; check hour meter to see if any scheduled maintenance is due, check the fuel level, check the engine oil level and fill as necessary to bring level to the upper dash mark on the dipstick, inspect the air cleaner for dents/cracks/loose connections, check tire pressure is between 85 and 100 pounds, check the fire extinguisher for security and readiness (to include that the pressure gauge indicates the proper range), check the fire suppression equipment for security and readiness for operation (to include looking for damaged tanks/hoses/other parts), inspect the operator compartment for cleanliness and wash out as required/check for damaged gauges and controls/operate all controls/test horn and all lights/verify all cables and linkage are clean and secure with no evidence of binding or sloppiness, and perform a general inspection to check the truck for any leaks/loose nuts and bolts and other damage to the truck with direction to correct or report any deficiencies to the service man. After starting the engine; monitor the transmission temperature as the engine warms to operating temperature (if the transmission temp exceeds 250° F, run engine at half-speed until the oil cools), check the ammeter and observe that the needle reads charge (+) and slowly returns to zero, observe that the engine oil pressure warning light goes off and that oil pressure is at least 30 psi at fast idle, continue to monitor gauges as the engine warms to operating temperature and observe that indications remain in the green zone, with the transmission in neutral reduce engine idle to half-speed and check the transmission oil level (add oil through the oil filler pipe as required to bring level up to the full mark), check the hydraulic oil level (with the dump box lowered and the oil at normal operating temperature 120° F) and fill as necessary to bring level up to the high mark, and check the parking brake by applying the brake and increase the throttle in second gear and service brake by rolling the truck forward and applying the brake to ensure the vehicle comes to an immediate stop.
- Every 125 hours or two weeks; perform all the 10 hour checks, wash the truck, check the (battery electrolyte level, differential oil level, front axle bolster rubber pads/bushings, and wheel end oil level), lubricate the (service brake pedal/1 fitting, center pivot ends/two fittings, throttle pedal/one fitting, steering cylinders/4 fittings, drivelines/16 fittings, front

axle bolster/1 fitting, dump box pivot pins/2 fittings, tailgate latch bar/2 fittings, dump cylinder pins/4 fittings, and tailgate pivot/2 fittings). Additionally, the fire suppressions system nozzle coverings and hose fittings are checked.

- Every 250 hours or monthly; perform all 125 hour items, check the air intake vacuum at 20" on a manometer, check the exhaust backpressure at 30" on a manometer, change the engine oil and dual filters, clean the air blower oil filter, check and adjust the engine valves (referring to Section 3 of the Deutz instruction manual), check the engine drive belts, clean the transmission breather, change the transmission oil filters, check the torque on wheel lug nuts at 450 feet/pounds, and check accumulator pressure at 900 psi.
- Every 500 hours or every two months; perform all 250 hour items, check the engine temperature gauges, check and clean the fuel injectors (including test of spray pattern), change the (fuel filters, transmission oil filters, and the hydraulic filters), check and clean the differential breathers, check the front and rear suspension (check bolt torque at 700 feet/pounds).
- Every 1,000 hours or every six months; perform all 500 hour items, change the air cleaner, check the alternator by testing the output, and change the differential oil/wheel end oil/hydraulic system fluid.
- Every 3,000 hours; have the complete fuel injection system inspected and serviced by a qualified diesel fuel system specialist.

WIPP performs the following preventive maintenance at the below specified intervals of equipment hours:

- Per the Underground Haulage Truck Equipment Operator qualification, the operator is trained to; check tire condition/inflation and lug nuts, check that the park and service brake are operational, check fuel/transmission/oil/hydraulic fluid levels, engine belt condition, readiness test for the fire suppression system, lights and horn are functional, check that the back-up and bed lower alarms are functional, and perform a walk-around inspection. Results are documented each shift on an Operator's Checklist.
- Every 100 hours of operation (per PM074061, Underground Diesel Mobile Equipment 100 Hour Inspection and Maintenance, Revision 8); the oil is changed, grease is applied to various components, the engine cooler/oil cooler/transmission cooler/boom/engine cooling fins around cylinders are cleaned using compressed air, the power train components are inspected for loose bolts/missing parts/oil leaks/motor mounts and the tires and wheels are checked, including torque lug nuts. Post maintenance testing specifically includes, "ENSURE proper oil level and NO oil leaks."
- Quarterly (per PM074027, Quarterly Diesel Emissions Test, Revision 8); test emissions for compliance with the WIPP Hazardous Waste Facility Permit.
- Every 500 hours of operation (per PM074080, EIMCO Haul Truck, Revision 3); the engine/cooling fins/oil cooler/transmission cooler/battery are cleaned using compressed air, the fan belts are checked for wear/cracks/gouges/tears as well as tension (0.28-0.35" deflection), inspect for loose bolts/missing parts/oil leaks, check and adjust tires for proper air pressure per the Operations and Maintenance Manual, check and restore water level in the batteries, check wheel lug bolt torque at 450 feet/pounds, change oil and filters, change

fuel filters, change hydraulic return filter, check and restore hydraulic fluid level, change transmission fluid filters, check and restore transmission fluid level, clean transmission breather, clean differential breathers, check and restore front/rear axle fluid level, and lubricate the (service brake pedal/one fitting, center pivot ends/two fittings, throttle pedal/one fitting, steering cylinders/4 fittings, drivelines/16 fittings, front axle bolster/1 fitting, dump box pivot pins/two fittings, tailgate latch bar/two fittings, dump cylinder pins/4 fittings, and tailgate pivot/two fittings).

- Every 1,000 hours of operation (per PM074080); perform all actions listed in the 500 hour maintenance, and sample the hydraulic fluid/transmission fluid/front & rear differential fluid/all four wheel end oils.
- Every 4,000-8,000 hours of operation; mechanical rebuild of the underground haul truck. Maintenance records of the 4,000 to 8,000 hour PMs were not provided to the Board.

The Board compared the WIPP preventive maintenance program for the salt haul trucks to the manufacturer recommendations and identified the following:

- The service manual prescribes several activities to be performed after starting the vehicle that were not listed on the Operations Checklist. Additionally, the Operations Checklist does not reflect all of the items listed in the Operations and Maintenance (O&M) Manual.
- Although the pre-operational checks are to be done referring to the O&M Manual, the items listed in the Underground Haulage Truck Equipment Operator qualification guide do not match the level of rigor identified in the O&M Manual.
- Operator's Checklists were not representative of the as-found condition of the underground vehicles.
    - On February 13, 2014, Salt Haul Truck 74U006A had active engine oil and hydraulic leaks observed by the Board that were not documented on the Operator's Checklist. The truck was taken out of service on the day of the event, but that action was due to a malfunctioning light.
    - Although provided on the checklist, restoration of fluid levels was not recorded.
- The service manual recommends washing the vehicle every 125 hours or two weeks and NWP accomplishes this task with compressed air.
- NWP performs activities prescribed for 125 and 250 hours at 100 hour intervals.
- The service manual recommends battery level be checked/restored at 125 hour intervals and NWP performs it every 500 hours.
- The service manual recommends inspection of the front axle bolster rubber pads and bushings every 125 hours and NWP performs it every 500 hours.
- The following recommended maintenance items listed in the Service Manual were not found in the NWP procedures:
    - Check the air intake vacuum at 20 inches on a manometer every 250 hours.
    - Check the exhaust back pressure at 30 inches on a manometer every 250 hours.

- Check and adjust the engine valves per the Deutz Instruction Manual every 250 hours.
- Check accumulator pressure at 900 psi every 250 hours.
- Clean the fuel injectors and test the spray pattern every 500 hours.
- Check front and rear suspension bolt torque at 700 feet/pounds every 500 hours.
- Check the alternator output every 1,000 hours.
- Change the wheel end oil and differential oil every 1,000 hours.
- Have the complete fuel injection system inspected and serviced by a qualified diesel fuel system specialist every 3,000 hours.

Additionally, during review of the service manual, the Board discovered that although the salt haul truck was built to use a fire resistant fluid in the hydraulic oil system, standard hydraulic fluid is used.

Corrective maintenance is initiated via submission of an Action Request (AR). The action request is screened, validated, and prioritized at the plan of the day meeting. If accepted, the scope is developed, an optimum work window is assigned, and the level of rigor in planning is determined to be minor maintenance, expedited work or planned work. Work planners, along with a planning team in some cases, further develop the activity level work control document and participate in development of a job hazard analysis.

The Board reviewed corrective maintenance records associated with the EIMCO Salt Haul Truck 74U006B. The following is a summary of corrective maintenance actions performed in the last ten years:

- Hydraulic Oil - 17 hydraulic oil leaks repaired since July 2004.
- Engine Oil - three oil leaks repaired since July 2004.
- Fuel System - four fuel-related leaks since July 2004.
- An insulating blanket was installed between the cab and the engine compartment to reduce the heat in the operator's compartment June 23, 2005.
- A Fire Investigation was performed on 74U006B following a fire on September 1, 2005.
- Electrical Repairs - 50 total (batteries, alternator, back-up alarm, headlights, taillights, horn and wiring repairs).

The engineer responsible for the haul truck and personnel in the maintenance organization were interviewed by the Board. Maintenance personnel indicated that the older haul trucks (74U006B, which was the truck involved in the fire, and an identical truck, 74U006A) were much more reliable and easier to work on than the newer haul trucks. Personnel also offered that equipment drivers prefer the newer haul trucks since they run cooler and ride more comfortably. When questioned regarding which underground equipment was more problematic, the interviewees indicated that the bolting machines were the main maintenance problem in the underground.

## 4.2 Salt Haul Truck Manual Onboard Fire Suppression System

Southwest Safety Specialists are under contract to perform maintenance of the manual onboard fire suppression system installed on the salt haul truck. Semiannually, the system undergoes a 25-step process to confirm that it conforms to National Fire Protection Association (NFPA) requirements. To date, there have been no significant anomalies identified.

## 4.3 Other Maintenance Related Issues

The Board visited the CMR and the underground, including the accident scene, on February 13 and 14, 2014. The following maintenance-related issues were identified:

- There was significant buildup of engine and hydraulic oil on other mining equipment including Salt Haul Truck 74U006A. (Figure 13)

**Figure 13: Buildup of Engine Fluids on the Underside of Vehicles in the Mine**

- There was a three-foot diameter puddle of hydraulic fluid underneath Salt Haul Truck 74U006A. (Figure 14)
- The daily Operator's Checklist was completed on February 5, 2014, for Salt Haul Truck 74U006A with no deficiencies indicated.
- There was an Out of Service tag on Salt Haul Truck 74U006A indicating that a lighting deficiency existed.
- Several mine phones were found to be inoperable (run to battery failure). Twelve of 40 phones tested were non-functional.
- Numerous components of the mine ventilation system were out of

**Figure 14: Hydraulic Fluid under Truck 74U006A**

service or otherwise impaired for an extended period of time, some since installation:

- Exhaust Fan 413, 41-B-700-A since January 27, 2014.
- Exhaust Fan 413, 41-B-700-B since June 15, 2013.
- 707 bulkhead door that divides the construction split from the disposal split requires manual operation and cannot be remotely shut, which is necessary for shifting to filtration mode. During the initial entries after the event, underground services shut the 707 bulkhead door and regulator the afternoon of February 14, 2014. This allowed the ventilation system to be placed in filtration mode. After the radiological event the evening of February 14, 2014, it would not have been possible to place the ventilation in filtration mode if 707 bulkhead door had remained open.
- 401 bulkhead door has been chained open for a long period of time. It cannot be operated remotely from the CMR in the chained condition. This is the bulkhead door from the Air Intake Shaft. See Figure 15 for an example.
- EXO regulator was not working. The garage door was opened about two feet, and allowed smoke in the EXO space. In its current configuration, this regulator cannot be remotely operated from the CMR.
- 504 bulkhead door was chained open for a long period of time. It cannot be operated remotely from the CMR in the chained condition. This is the bulkhead door to the Salt Handling Shaft.
- 308 bulkhead regulator cannot be remotely operated from the CMR due to the regulator being out of service or impaired. This bulkhead is located between the Waste Shaft and the exhaust shaft.

**Figure 15: One of Chained Bulkhead Doors**

Numerous other pieces of equipment were out of service or otherwise impaired:

- 534-CAM-001-152 has only been operational a total of 29 days in the last 22 months.
- Building 463 Compressor Building trouble alarm has not been energized since May 21, 2013.
- Area 451 CMR Fire Alarm Panel impaired since June 5, 2013.
- Building 486 Northeast site, Riser Flow Switch Valve closed due to system leaking on August 9, 2013.
- Hydrant #23 out of service since September 10, 2013.
- Hydrant #3 out of service due to no flowing water since September 16, 2013.
- Auxiliary Warehouse FAP Broken Pull Station since October 27, 2013.
- Fire Water PIV #FW-Y-PIV-21 unable to operate in the closed direction since December 23, 2013.
- Fire Water PIV #FW-Y-PIV-27 is shut to isolate Hydrant #5 due to leakage since December 30, 2013.
- Fire Panel 031 not sending alarm signal to CMR since January 6, 2014.
- Gate House fire panel light going out since January 28, 2014.

Additionally witness statements and interviews from personnel yielded the following:

- PA announcements were difficult to hear or understand.
- There is a difference in expectations for waste-handling vs non-waste-handling vehicles.
- Pre-operational checks are not identifying equipment problems that need to be addressed other than light and horn issues.
- Some mine phones were reported as not working properly or difficulty in hearing was experienced.
- Thirty-three emergency lights in the Waste Handling Building have been inoperable for as long as two years.

**Analysis**

The Board determined that the use of fire resistant fluid in the hydraulic system could have significantly reduced the quantity of combustible liquid on the haul truck. Additionally, rigorous inspections and policing of oil and grease accumulation could have further reduced the combustible loading on the haul truck.

The Board determined that there is a significant delta between the preventive maintenance prescribed in the service manual and what is performed. Routine monitoring and adjustments that are not included in the NWP procedures are important maintenance items that could affect engine performance, resulting in higher than normal operating temperatures. Additionally, several decisions regarding maintenance and upkeep of the salt haul trucks were made without sound engineering judgment and evaluation. Discontinuing use of the wash station and opting

for compressed air as the means to keep the vehicle clean significantly inhibits the ability of maintenance personnel to identify and correct fluid leaks, resulting in continued buildup of combustibles.

The Board reviewed the equipment status and condition in the CMR and the U/G. The condition of critical pieces of equipment, such as the 700 exhaust fans, indicates that management has not taken prompt action to resolve longstanding deficiencies. Many items have been out of service or in a reduced status for more than six months. It was not clear that NWP had a clear approach to prioritizing maintenance activities in regard to critical equipment or that there is an effective formal process to identify compensatory measures other than a fire watch for impaired safety-related equipment. Additionally, the equipment and components that affect normal operation of the mine ventilation system did not appear to have been effectively evaluated and dispositioned regarding their impact on system operation. (Figure 16)

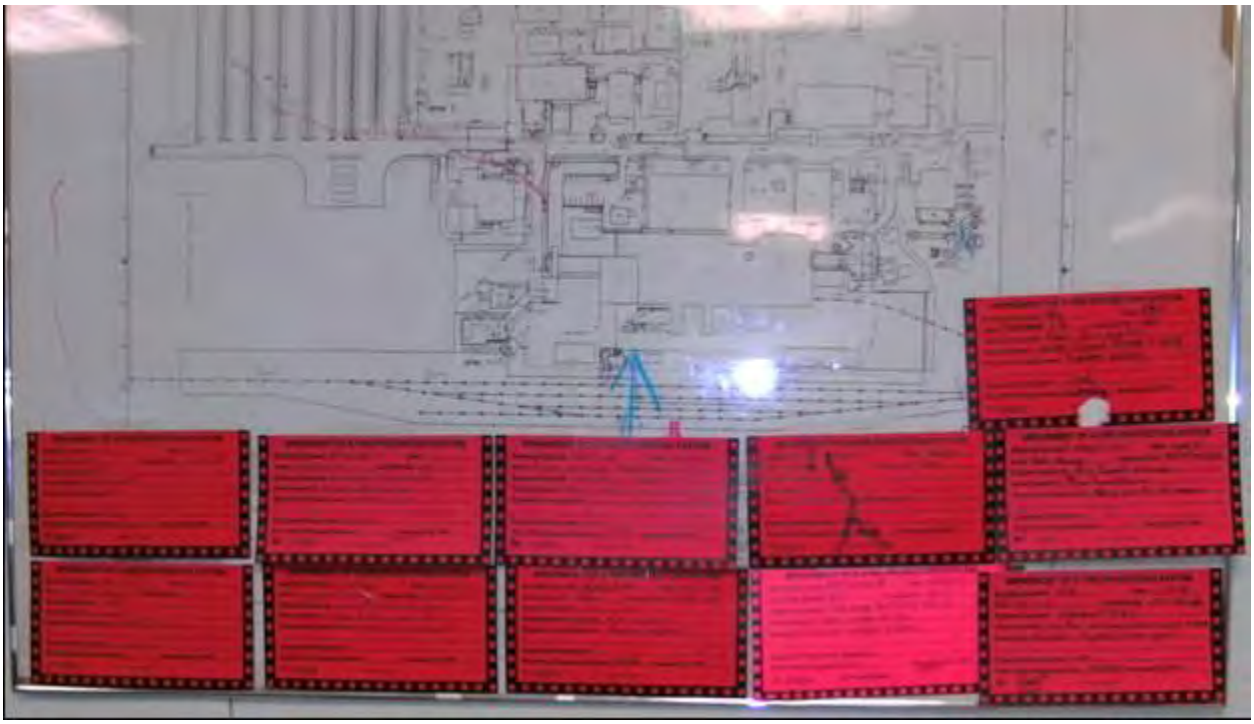


**Figure 16: Fire Protection System Impairment (Out-of-Service Tags) in the CMR**

**CON 6**: The NWP preventive and corrective maintenance program did not prevent or correct the buildup of combustible fluids on the salt haul truck.

**JON 13**: NWP management needs to reevaluate and modify the approach to conducting preventative and corrective maintenance on all underground (U/G) vehicles such that combustible fluids are effectively managed to prevent the recurrence of fires.

**CON 7**: NWP and CBFO management is not adequately considering overall facility impact with regard to operations, emergency response, and maintenance, which affects the safety posture of the facility, e.g., salt haul truck combustible build-up, conversion of the automatic fire suppression system to manual, removal of the automatic fire detection capability, not using fire resistant hydraulic fluid, discontinued use of the vehicle wash station, chaining of ventilation doors and an out-of-service regulator and fans, inoperable mine phones, and other non-waste-handling related equipment.

**JON 14**: NWP and CBFO need to develop and implement a rigorous process that effectively evaluates:

- changes to facilities, equipment, and operations for their impact on safety, e.g., plant operations review process;
- impairment and corresponding compensatory measures on safety-related equipment; and
- the impact of different approaches in maintaining waste-handling and non-waste-handling equipment.

**JON 15**: NWP needs to determine the extent of this condition and develop a comprehensive corrective action plan to address identified deficiencies.

**CON 8**: NWP and CBFO management have not effectively managed the quantity and duration of out-of-service equipment.

**JON 16**: NWP needs to develop and implement a process that ensures comprehensive and timely impact evaluation and correction of impaired or out-of-service equipment.

**JON 17**: CBFO needs to ensure that its contractor oversight structure includes elements for comprehensive and timely evaluation and correction of impaired or out-of-service equipment.

**CON 9**: NWP management has allowed less than acceptable rigor in the performance of equipment inspections allowing the operation of U/G equipment in unacceptable condition.

**JON 18**: NWP needs to develop and reinforce clear expectations regarding the performance of rigorous equipment inspections in accordance with manufacturer recommendations, established technical requirements; corrective action; and trending of deficiencies.

# 5.0 Fire Protection Program

Understanding fire hazards is essential to risk reduction and fire protection decision-making. DOE O 420.1C, *Fire Protection*, requires a documented fire protection program including comprehensive, written fire protection criteria or procedures, fire hazards analysis (FHA) and a baseline needs assessment (BNA) of the fire protection emergency response organization.

An FHA is a tool used to understand fire hazards. The process of quantifying the fire hazard is driven by the need to determine the overall hazard of a process or facility or to have a decision-making tool for fire protection systems. An FHA is an important element of risk assessment and can also be used as a stand-alone hazard evaluation tool.

The benefits of conducting an FHA include:

- An inventory of fire hazards, including quantities.
- A comprehensive understanding of the fire hazard, including potential magnitude and duration.
- An estimate of the potential impact of a fire on personnel, equipment, the community, and the environment.
- Development of a list of appropriate mitigation options.

A BNA establishes the site firefighting capabilities necessary to suppress all fires. It also establishes the necessary emergency medical and hazardous materials response capabilities. This includes an evaluation of staffing, apparatus, facilities, equipment, training, preplans, offsite assistance, and procedures.

The Board reviewed the fire protection program with a focus on the implementation of requirements documented in the FHA, BNA and requirements related to the combustible material control program. The FHA is documented in WIPP-023, *Fire Hazard Analysis for the Waste Isolation Pilot Plant*, Rev. 6. The BNA is documented in DOE/WIPP-11-3471, Rev. 1.

## 5.1 Fire Hazard Analysis

The FHA indicates that it is implementing the requirements of DOE O 420.1C, and DOE-STD-1066-12, *Fire Protection*. The FHA evaluates fire in the following sections:

- Underground Disposal Circuit (5.2.2),
- Underground Construction Circuit (5.2.3),
- Underground North Circuit (5.2.4), and
- Common Facility Fire Scenarios (5.2.5).

The Board found the following in the FHA:

- The FHA does not address the possibility that the vehicle fire suppression system does not perform as intended. The FHA does not consider the possibility that the onboard vehicle fire suppression system could fail to extinguish the vehicle fire.
- The FHA addresses a fire near the air intake on the surface, but does not consider the smoke/products of combustion migration throughout the underground if the fire is in the air intake drift.
- The analysis of a vehicle fire does not differentiate the level of protection provided by a manual fire suppression system versus the level of protection provided by an automatic fire suppression system.
- Life Safety for the Shafts and Underground (7.6) is not evaluated using the same criteria as all other facilities at WIPP. The above ground facilities use DOE O 420.1C and NFPA 101, *Life Safety Code,* versus MSHA requirements used in the underground. The FHA references the 1998 version of 30 CFR 57, *Safety and Health Standards*, "Underground Metal and Non-metal Mines" (MSHA).
- The reflectors intended to mark the worker egress evacuation direction were difficult to see during the evacuation of the underground. This was a concern for several personnel as they evacuated the underground.
- The FHA does not address how omitting automatic fire suppression systems from the underground and its various vehicles and enclosures meet the requirement of DOE O 420.1C (Attachment 2, Chapter II, Section 3.c.(2)(b)).
- Additionally, the Safety Class fire systems (4.3.3) on the waste haulers are not designed to meet single-point failure criteria.

The DOE-STD-2012, *Fire Protection*, Appendix B, Section B.25 states the FHA should evaluate the consequence of a single worst case automatic fire system malfunction.

**Analysis**

The FHA does not provide a comprehensive analysis that addresses all credible underground fire scenarios, including a fire located in the air intake drift. The FHA did not consider the ventilation system movement of smoke/products of combustion throughout the mine. Additionally, the FHA analysis of vehicle fires is insufficient to provide an advance understanding of potential impacts or necessary mitigative actions associated with this or other potential vehicle fires. The FHA fails to address the impacts caused by the difference in the level of protection provided by manual versus automatic detection and fire suppression systems.

DOE O 420.1C requires an FHA to be prepared for nuclear facilities and for facilities with unique hazards. The underground meets both of these criteria. The FHA does not identify, discuss, evaluate, and analyze the unique hazards in the underground. The FHA must describe the controls necessary to address these unique hazards.

The Documented Safety Analysis has identified the waste-handling vehicles' fire suppression system as safety class; however, the FHA fails to identify how these systems are protected

against single-point failures. In addition, these systems are focused risk reduction tools that address specific vehicle fire scenarios. They are not comprehensive protection systems equivalent to automatic sprinkler systems in buildings. DOE facilities have historically credited manual intervention and detection that automatically notifies a response organization for protection against single-point failures. Since there is not a fully defined response organization to fight fires in the underground, the FHA needs to identify the suppression system that is required.

The FHA discussion on life safety does not include a reference to a DOE-approved exemption/equivalency for application of the MSHA requirement instead of the NFPA 101. The FHA implies the MSHA requirements provide an equivalent level of protection without objective evidence to support the assumption. Objective evidence in the form of an approved exemption/equivalency for meeting the DOE Fire Protection program requirements must be established.

A lack of thorough analysis and development of the fire program requirements resulted in a lack of adequate information to ensure risk-informed, conservative decision making could be applied with regard to the fire protection program.

## 5.2 Baseline Needs Assessment

The BNA and the status of recommendations from the BNA were reviewed and the following items were identified:

- BNA Recommendation 2012-10 states: "WP 12-ER4911 does not define minimum response, response roles or resource capabilities. It is recommended to amend this procedure to more clearly define such things as FLIRT response actions and Rescue Truck #2 response (page 32)." This issue is from the 2010 version of the BNA and was not resolved in the 2012 revision to the BNA.

  > *"The MRT is not dispatched to fight fires in the U/G. They will be activated if the fire is beyond the incipient stage and search, rescue and/or recovery operations are needed. They will engage in firefighting only as necessary to carry out rescue operations."* BNA, page 35

- Underground Fire Response analysis in section 7.3.1.2 states:

  > *"U/G fire response is documented in WP12-ER4911, U/G Fire Response. Workers discovering fire in the U/G are trained to contact the CMR. Workers are expected to evaluate and respond to incipient fires with portable fire extinguishers. If the fire is vehicle-related, initial U/G fire response is to use automatic or manual vehicle fire suppression systems.*
  >
  > *The CMR will contact Underground Services personnel who will make an evaluation of the fire. Based on that evaluation, Underground Services will extinguish the incipient stage fire with a portable fire extinguisher or initiate U/G evacuation through the CMR Operator. The CMR will make an announcement informing personnel of the fire location, instructing personnel in smoke to don self-rescuer, suspending all U/G operations, and instructing to U/G personnel report to egress hoist stations. Per the*

*direction of the CMR and U/G Services, Emergency responders or FLIRT members will respond to S700/E140 to bring Rescue Truck #2 to the incident. Once evacuation is complete, a response plan is developed depending upon the status of the fire. The plan may include ventilation control, barrier erection, and waiting for the fire to self-extinguish or implement active ventilation."* (p.32)

- 9.1 Existing Recommendations states that:

  *"The communicator paging system is old and needs to be updated or replaced. The old system has been replaced. The new system was placed in service in August 2012.* Status: "Completed in August 2012"

- Recommendation 2012-10 states: "Define minimum response and response capability in procedure WP12-ER4911."
- Recommendation 2012-10 Supporting Statement states:

  *"The U/G fire response procedure does not define minimum response, response roles or resource capabilities, such as FLIRT actions to be taken, nor does it outline deployment of possible resources, Rescue Truck #2 and the 300-pound wheeled ABC fire extinguisher.*

  *During review of documentation, the response provided by Rescue Truck #2 and the 300-pound wheeled ABC fire extinguisher was not evident. Rescue Truck #2 contains an onboard 150-pound foam extinguisher and 125-pound dry chemical extinguisher. Documentation was not found to indicate who is authorized to use them nor is intended use specified. If these resources are proven to not be value added, then recommend removing them from the U/G to prevent confusion or misuse." Page 62*

- Hazardous Material and Radiological Event Responses, section 7.3.2 states**:**

  *"For a fire that may damage TRU waste containers or radioactive sources, the CMR sounds an alarm and makes an announcement for all personnel in the affected area to evacuate to an area with clean air to await Radiological Control Technician (RCT) arrival. The CMR will activate the FB. If the event occurred U/G, ventilation will be adjusted to ensure negative differential pressure in the affected areas and verify the high-efficiency particulate air filter bank differential pressures are normal."*

The contractor is required to provide emergency response capabilities, as necessary, to meet site needs as established by the BNA, safety basis requirements, and applicable regulations, codes and standards as required by DOE O 420.1C, section 3.e. Evidence to support implementation of the above recommendations could not be found. These recommendations, some of them going back to 2010, remain unresolved and unimplemented.

The audibility of the communicator paging system was a concern for some of the personnel evacuating the underground. It has been noted that some of the old amplifiers are still installed in the communicator paging system, although the BNA identifies that the old communicator paging system has been updated.

There is no formal documentation (e.g., equivalency or exemption) describing the alternative method for ensuring the safe egress of underground personnel and how the alternate method fulfills the requirements of DOE O420.1C, 3.c.1.

If relying on manual fire suppression, DOE O 420.1C, Attachment 2, Chapter II, section 3.e. (1)(a) requires pre-incident strategies, plans, and standard operating procedures to be established to enhance the effectiveness of manual fire suppression activities. The existing procedures do not address this.

The CBFO approved the BNA without comment regarding the longstanding open recommendations.

The BNA indicates that fighting anything past an incipient stage fire in the underground is only done by the MRT. The MRT only fights fires to support rescue of personnel, not to protect property. The BNA should be updated to reflect the actual MRT approach to limit fire damage, but only after the underground is fully evacuated. Also, the MRT will typically avoid direct suppression of a fully developed fire, and instead erect barriers from a safe location that directs ventilation away from fires. The effect of this approach limits fire damage while the fire self-extinguishes by consumption of fuel.

Workers evacuating the underground were confused by the shift in ventilation mode, adding stress to the existing emergency condition.

**Analysis**

The BNA has not met one of its basic functions, determination of the current and future needs for the emergency service aspects of fire suppression in the underground. It does not determine the minimum manpower, equipment and training needed to manage a fire in the underground. Instead it assigns a recommendation to "Define minimum response and response capability in procedure WP12-ER4911." Assigning the task to evaluate the minimum response and response capabilities to an operations procedure puts an undue burden on the procedure writer. The BNA must indicate training, staffing and equipment necessary for safe operations. The implementing procedure can then address how to make the best use of the defined staffing as established in the BNA. The BNA must be developed to identify the necessary requirements and flow those requirements into the implementing procedures.

The BNA section 7.3.1.2 allows for adjustment of ventilation without any analysis of the effect of a ventilation change or under what circumstances an adjustment is inappropriate. The BNA should incorporate an analysis and determine the appropriate limitations on the use of ventilation system changes in the event of an underground fire.

The BNA closed out a past recommendation concerning the paging system without the new installation being completed. It clearly states the paging system replacement was completed in

August 2013. However, the facts indicate old amplifiers are still installed and have not been replaced. The recommendation needs to be reopened and closure needs to be validated by NWP and verified by CBFO to ensure the new system is actually capable of performing its intended function.

The WIPP facility needs to embrace its dual nature of being a mine as well as a Hazard Category 2 Facility. As such, WIPP has two distinct requirement sets, MSHA and DOE O 420.1C. Both of these have fire protection program requirements that must be met. There is a common misconception that MSHA is the only program requirements for underground operations. Both sets of requirements must be met and any deviation fully addressed. Therefore, NWP needs to perform a line by line review of DOE O 420.1C requirements (Attachment II, Chapter 2) and MSHA requirements to ensure both requirement sets are fulfilled. Where differences exist, they need to be identified, evaluated and reconciled properly. This is not limited to just the evaluation of automatic suppression and Life Safety code. It should also include emergency response requirements for the underground, including strategies and preplans for fire events.

## 5.3 Underground Combustible Material Storage

Good housekeeping and control of combustible/ignition sources are basic components of any fire protection program (FPP). External reviews from the DNFSB have identified a long-standing issue with the control of combustible materials at WIPP. Additionally, an EM HQ assist-visit noted that "Combustible loading limits establish safe storage arrangements for spools of wire and combustible materials in the underground. Current conditions far exceed those limits. The FPP has not been adequately assessing the combustible material limits established for the underground."

**Figure 17: Combustible Loading in the Mine**

While these fire protection reviews addressed only the control of combustible materials, material storage and staging is really the issue. Salt movement requires periodic touchup to the ribs and back (walls and ceiling) to maintain the non-waste work areas, and the floors throughout the mine, in good repair. As a result, equipment and materials are moved out of an area and staged in the drifts. Although this staging is temporary storage, the condition can last several months. The Board observed materials

stored on either side of the drifts, materials that obscured reflectors, and stored combustibles exceeding the 5 megawatt (MW) limit. Personnel group interviews indicated storage on only one side of drift could have made navigation in drift easier.

This 5 MW limit has not been enforced and NWP is not in compliance with the limits. The use of office furniture in some areas will exceed the 5 MW limit. There were numerous examples of accumulation of combustible materials in the underground that exceed either the spacing requirement or the 5 MW accumulation limit, as evidenced in Figure 17.

**Analysis**

NWP and CBFO have allowed for the use and accumulation of combustible material in the U/G in excess of the limits allowed by the fire analysis and implementing procedures. NWP does not appear to practice an "As Low As Reasonably Achievable" (ALARA) posture regarding the use and storage of combustible materials in the underground.

**CON 10:** NWP did not ensure the BNA addressed requirements of DOE O 420.1C and MSHA with the results completely incorporated into implementing procedures.

**JON 19**: NWP needs to ensure that all requirements of DOE O 420.1C and MSHA are addressed in the BNA with the results completely incorporated into implementing procedures and the source requirements referenced, and that training consistent with those procedures is performed.

**CON 11:** NWP and CBFO management did not make conservative or risk-informed decisions with respect to developing and implementing the fire protection program.

There is inadequate fire engineering analysis due to a lack of integration with ventilation design and operations, and U/G operations, for recognizing, controlling, and mitigating U/G fires.

**JON 20**: NWP and CBFO need to perform an integrated analysis of credible U/G fire scenarios and develop corresponding response actions that comply with DOE and MSHA requirements.

The analysis needs to include formal disposition regarding the installation of an automatic fire suppression system in the mine.

**CON 12**: NWP and CBFO have failed to take appropriate action to correct combustible loading issues that were identified in previous internal and external reviews.

**JON 21**: NWP and CBFO need to review the combustible control program and complete corrective actions that demonstrate compliance with program requirements. These issues remain unresolved from prior internal and external reviews.

**CON 13**: NWP and CBFO have allowed housekeeping to degrade and other conditions to persist that potentially impede egress.

**JON 22**: NWP and CBFO need to evaluate and address deficiencies in housekeeping to ensure unobstructed egress and clear visibility of emergency egress strobes, reflectors, SCSR lights, etc.

## 5.4 Fire Forensics

This fire description has been prepared based on a partial visual inspection conducted on February 13, 2014, Board interviews, evaluation of numerous photos taken on February 13, 2014, and other available data. Further inspection was prevented by a radiological contamination event that occurred on February 14, 2014.

Fire ignition is presumed to have occurred near the exhaust system piping on the lower left side of the haul truck forward of the front wheel (see Figure 18). This area is enclosed by steel construction, making early visual detection difficult. The initial material ignited was a combustible liquid leaking onto the exhaust system. The liquid could have been free flowing or an accumulation on exposed surfaces. The flashpoint temperatures for haul truck fluids are listed in Table 2. The operating temperature of a typical catalytic converter will range from 300 to 500° C; however, this may exceed 500° C during abnormal engine operation (NFPA 921-2014, *Fire and Explosion Investigations*). All of these values exceed the flashpoints shown in Table 2.

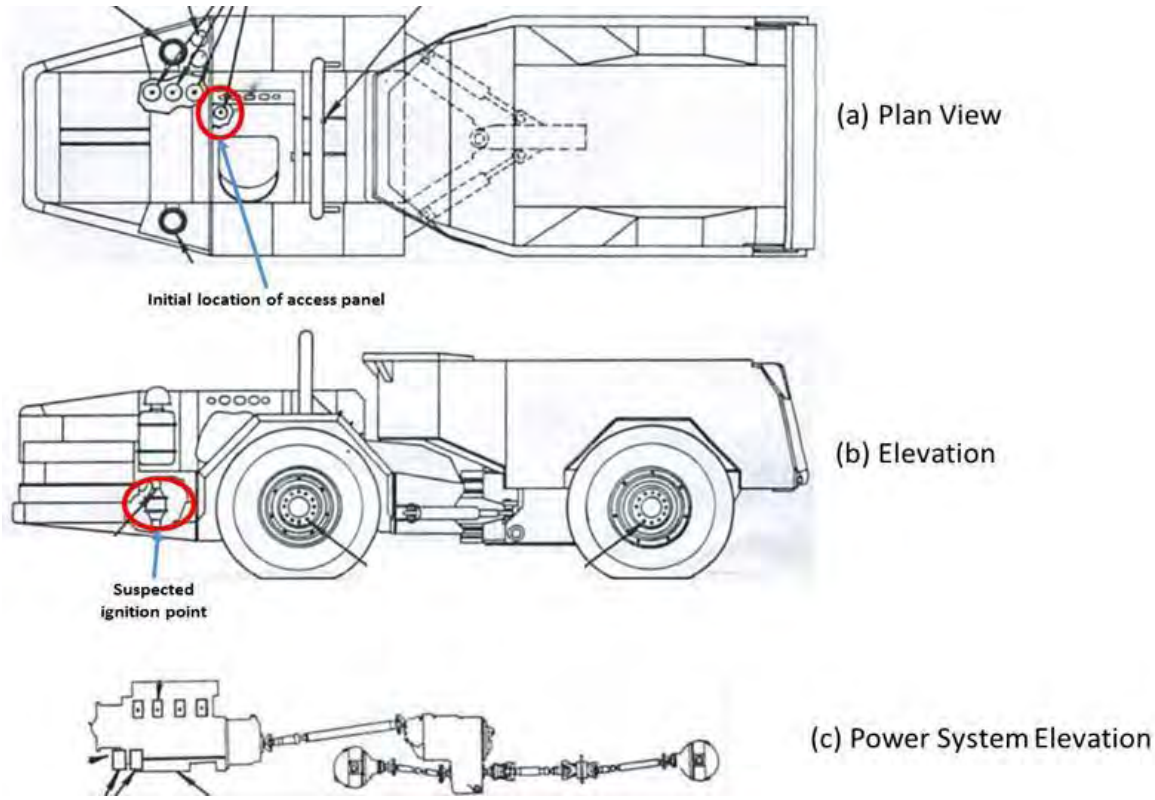



**Figure 18: Salt Haul Truck**

**Table 2: Liquid Fuels on Salt Haul Truck**

| System | Product | Flashpoint °C |
|---|---|---|
| Engine | Exxon Mobil XD-3 30 Oil | 220 |
| Hydraulic | Citgo A/W Hydraulic Oil 68 | 242 |
| Transaxial differential | Citgo Regular Gear Oil, SAE 90 | 236 |
| Transmission | Citgo Transgard® MP ATF | 208 |

**Figure 19: Engine Cooling Coils**

The Board observed accumulated grime in the engine compartment of other salt haul vehicles that would have been sufficient to create a sustained fire that would result in additional fluid leaks. The normal hauler ventilation flow pulls air through the front grill and directs it across the cooling coils above the engine block (see Figure 19). This flow exits at the back of the engine compartment and would push the fire towards the transmission (see Figure 18, illustration (c)). In the actual fire, this condition occurred within about 5 to 10 minutes of initial ignition (assumed start time is 1043) and was recognized by the truck Operator at 1048. The Operator shut down the engine, exited the hauler on the left side, and moved to the right side. The Operator then discharged his hand-held fire extinguisher through the transmission access hole, which is forward of the pivot pin (see Figure 20) and under the haul truck. Because the fire was enclosed, using a handheld fire extinguisher was ineffective. The delay between initial ignition and activation of the engine compartment fire suppression system resulted in development of multiple leaks.

**Figure 20: Transmission Fluid Stick Access**

When the Operator had discharged the hand-held fire extinguisher, he returned to the left side of the haul truck to initiate release of the on-board fire suppression system. The effectiveness of suppression system is uncertain. At activation, the fire could have been too severe to control, or if initially controlled, a hot surface within the engine compartment might

have reignited the fire. When the engine was shut down, the cooling fan stopped. Because of the haul truck orientation, mine ventilation flow was opposed to the cooling system airflow. This prevented further propagation towards the rear of the haul truck and prevented ignition of the rear tires. The change in engine compartment airflow direction also created a V-shaped discoloration on the haul truck grill. This discoloration was created by the most intense flaming. Sooty deposits occurred on either side of this V where the flames were less intense.

Ignition of the front tires likely occurred within 10 minutes of fire ignition (see Figure 21). Involvement of the tires produced heavy black smoke at the Salt Handling Shaft station before 1053 (1051 plus two minutes for the Salt Handling Shaft Bottom Lander to move to the station).

**Figure 21: Salt Haul Truck Damage**
**(**Engine Cowling Was Opened Post-Fire**)**

The majority of the airflow entered the underground via the Air Intake Shaft, moved by the flaming salt haul truck, and moved towards the Salt Handling Shaft. This airflow, which moved from the back to the front of the truck, created a well-ventilated fire within the truck. Flames from the combustible liquid and tires impinged on the salt rib and caused spalling of the salt (see Figure 22). A smoke signature carried from this impingement point to the bulkhead doors at the base of the Salt Handling Shaft (see Figure 23). The doors created a well-mixed flow at the Salt Handling Shaft. The Exhaust Shaft flow pulled the well-mixed combustion products into the Waste and Construction Air Handling Circuits. The elevated local temperature also created an upcast flow through the Salt Handling Shaft.

**Figure 22: Haul Truck and Rib Spalling**

**Figure 23: Smoke Signature on Rib (Looking South)**

A portion of the airflow entering the underground traveled through the North Air Handling Circuit. The salt haul truck fire created a smoke layer in this circuit. This layer was approximately two feet deep, and traveled to the partially opened rollup door in E140. This arrangement trapped the upper smoke layer since the lower layer moved through the door.

At 1058, the FSM initiated a reduction of airflow; however, the fire remained well-ventilated. The change significantly decreased airflow through the North Air Handling Circuit and permitted combustion products to drop to the floor (see Figure 24). Combustion products continued to travel to the Salt Handling Shaft station to be pulled into the Waste and Construction Air Handling Circuits by the Exhaust Shaft flow, or to be upcast through the Salt Handling Shaft.

**Figure 24: Soot Deposits in North Ventilation Circuit**

Sometime during the intense burning period an accumulator within the engine compartment burst. This ejected an end-cap which ruptured an access plate and went through the hauler operator compartment. The end-cap traveled approximately 10 feet beyond the back of the haul truck (see Figure 25). The access plate was severely deformed and traveled approximately 20 feet beyond the end of the haul truck (see Figure 26).


**Figure 25: Accumulator Endcap (MG 3591)**

**Figure 26: Damaged Access Plate**

Significant combustion continued for 20 to 40 minutes (Time 1103 to 1123). Underground evacuation continued until 1134. The fire continued to smolder until the Mine Rescue Team applied foam fire suppressant at 2300.

The above analysis is consistent with the report from Investigator Robert Brader, attached in Appendix F.

# 6.0 Safety Equipment

The Board reviewed safety equipment to determine the impact on the event. The salt haul truck fire suppression system, emergency breathing equipment, underground ventilation, and the U/G communication and emergency notification systems were evaluated.

## 6.1 Salt Haul Truck Fire Suppression System

### 6.1.1 System Description

The fire suppression system installed on the salt haul truck (Vehicle 74-U-06B) is an ANSUL A-101-30 Dry Chemical Fire Suppression System that contains 30 pounds of FORAY dry chemical agent for Class A, B, and C fires. The ANSUL A-101 Fire Suppression System is a Factory Mutual (FM) approved pre-engineered, cartridge-operated dry chemical system with a fixed nozzle distribution network designed for use on large, off-road type construction and mining equipment, underground mining equipment and specialty vehicles.

The system is released manually by activation of one of two mushroom buttons (pneumatic actuator) located on the front wheel fenders. When pushed by the vehicle operator (or an observer) the pneumatic actuator ruptures a seal disc in the expellant gas cartridge. This, in turn, pressurizes and fluidizes the dry chemical extinguishing agent in the tank, ruptures the burst disc when the required pressure is reached, and propels the dry chemical through the network of distribution hose. The dry chemical is discharged through fixed nozzles and into the protected areas, suppressing the fire. According to the Southwest Fire Safety Company, responsible for maintaining this system for the past 19 years, there are six nozzles, four in the engine compartment and two in the transmission compartment where the fire was first observed.

There were no design drawings for the system provided. Physical verification of the complete system configuration was not possible due to inability to reenter the mine.

### 6.1.2 System Configuration

The salt haul truck was originally procured without a fire suppression system. The ANSUL A101 system was originally installed on vehicle 74-U-006B at an unknown time, but records indicate that it was prior to 1995. The following items identify activities that affected the fire system:

1. The system was recharged in April of 2000 after discharge.

2. The automatic suppression system defeat switch was removed May of 2000. The vendor responsible for fire suppression system maintenance indicated that this switch's function was to delay the automatic discharge of the dry chemical.

3. The system was changed from automatic to manual operations on October 21, 2003, via Work Order ID 0300900, created January 28, 2003. This transition included replacing the actuator with new A-101 actuator with accessories. The new system configuration did not include automatic detection or automatic engine shutdown. Both of these were functions from the original installation.

4. After investigation of the haul truck fire of September 2005, a subsequent work order was executed to replace a battery cable that was damaged in the fire. No other damage was cited. A PM done on April 17, 2006, that included providing 30 pounds of dry chemical agent.

**Analysis**

A vehicle fire suppression system is designed to suppress a fire and reduce fire size and heat output, but not necessarily extinguish all fires. The onboard fire extinguisher should be used to extinguish residual small fires remaining after system discharge.

The manual system is only discharged when an operator takes two conscious actions: pull the pin and push the actuator. Vehicle shutdown is an additional step that is necessary to remove the engine heat to ensure extinguishment of the fire. The automatic system contained detection and automatic vehicle shutdown capability that would not require human intervention.

The combination of the operator using a hand held extinguisher before initiating the manual fire suppression system provides an example of why the automatic system is the preferred approach. The delay in activation of the manual system is likely to allow the fire to grow beyond the incipient stage by the time it is detected by the truck operator. Automatic detection and extinguishment is preferred. The impact of switching the suppression system from automatic detection and activation to manual activation modes was not fully analyzed.

**CON 4**: Actions to be taken by the Operator in the event of a U/G vehicle fire were not clear.

There were inconsistencies between procedures and training for fire response that led to an ineffective response to the salt haul truck fire.

**JON 8**: NWP needs to review procedures and ensure consistent actions are taken in response to a fire in the U/G.

**JON 9**: NWP, CBFO and DOE HQ need to clearly define expectations for responding to fires in the U/G, including incipient and beyond incipient stage fires.

## 6.2 Emergency Breathing Equipment

### 6.2.1 Description of Self-Rescue and Self-Contained Self-Rescue Devices Underground (Manufacturer)

**W-65 Self Rescuer**: The W-65 Self Rescuer is designed to protect the wearer from carbon monoxide only and will support the user for 60 minutes in a carbon monoxide environment. This device should not be used in atmospheres containing less than 19.5 percent oxygen. The W-65 Self Rescuer is belt worn and should never be farther than arms reach away from the person it is assigned to. Under no circumstances should the distance from the employee and the W-65 Self Rescuer ever exceed 25 feet. It is to be used in the event of a fire or smoke for emergency egress or to get to a cache of Self Contained Self Rescuers (SCSR's).

**OCENCO EBA 6.5 Self Contained Self Rescuers (SCSR's):** The OCENCO EBA 6.5 has been approved as a 1 hour closed circuit self-contained self-rescuer. Extensive testing has shown that the OCENCO EBA 6.5 will provide the user with over 60 minutes of life saving oxygen in escape situations requiring heavy physical activity. There are 425 of these units at storage locations throughout the underground facilities.

Similarly, the OCENCO EBA 6.5 has demonstrated the ability to provide the user with up to 8 hours of oxygen if he remains at rest and follows the procedures necessary for maximum conservation as explained in the Instructions for Use for Users at Rest.

MSHA requires that all persons who may depend on the OCENCO EBA 6.5 for survival be thoroughly trained in the operation and use of the unit.

**FireHawk M7 Air Masks**: The FireHawk M7 Air Masks are pressure-demand, self-contained breathing apparatus (SCBA) certified by the National Institute for Occupational Safety and Health (NIOSH) for use in atmospheres immediately dangerous to life or health (IDLH).

This Air Mask complies with the National Fire Protection Association (NFPA) for Open-Circuit Self-Contained Breathing Apparatus for Fire Fighters. The Air Mask will protect the user from CBRN (chemical, biological, radiological, and nuclear). Four of these units are stored underground at the WIPP mine. They can be found on the Emergency Rescue Wagon located at S-700 and E-140. These units are for Fire Fighting Use in the event of an Emergency only by specifically trained and qualified personnel.

Both the W-65 Self Rescuers and the SCSRs were used by underground workers during the evacuation from the mine during the haul truck fire incident. The Board has reviewed the statements of workers who were in the mine at the time of the incident. The Board was able to look at the results of the evacuation on documents from 61 of the 86 workers that successfully escaped the haul truck fire.

Six employees (10 percent) of those who provided documentation did not use the W-65 Self Rescuer. Three of the six employees did not use a self-rescuer at all during the evacuation. The other three used the SCSR. Fifty-five of the employees performed as trained and donned their W-65 Self Rescuers (90 percent).

- Four of the 61 had difficulty opening the W-65;
- Thirteen of the employees were able to successfully use the SCSR; and
- Twenty one of the SCSRs did not open properly and could not to be used.

**Analysis**

Many individuals had difficulty donning either the SCSR or W-65 self-rescuer. There is no training that simulates use in likely emergency conditions (i.e., limited visibility due to dark or smoke filled areas). The annual refresher is a video that does not require donning of the SCSR. It is at the trainee's discretion whether or not they desire to don the SCSR or W-65 during training. The existing training program for use of the SCSR and W-65 self-rescuer does not evaluate the competency of the user.

**CON 5**: NWP and CBFO failed to ensure that training and drills effectively exercised all elements of emergency response to include practical demonstration of competence, e.g., donning of self-rescuers and SCSRs, U/G personnel response to a fire, use of portable fire extinguishers, EOC roles, classification and categorization, notifications and reporting, and allowance of unescorted access for over 500 personnel, etc.

**JON 10**: NWP and CBFO need to develop and implement a training program that includes hands-on training in the use of personal safety equipment, e.g., self-rescuers, SCSRs, portable fire extinguishers, etc.

**JON 11**: NWP and CBFO need to improve and implement an integrated drill and exercise program that includes all elements of the ICS, including the MRT, First Line Initial Response Team (FLIRT) and mutual aid; unannounced drills and exercises; donning of self-rescuers/SCSRs; and full evacuation of the U/G.

## 6.3 WIPP Underground Mine Ventilation

The underground ventilation system (UVS) serves all underground facilities and provides the equipment, controls, and monitoring necessary to provide a suitable environment for underground personnel and equipment during normal activities. It also provides confinement and channeling of potential airborne radioactive material in the event of an accidental release or smoke and fumes in the event of an underground fire. It further provides high-efficiency particulate air (HEPA) filtration of exhaust air to minimize any doses to onsite and offsite personnel. Under normal operating conditions, the effluent exhaust is not filtered. The status of the system equipment is continuously monitored, and the data are provided to the CMR, as well as local stations underground.

The air is supplied to the underground, at 2,150 feet below the surface, through three shafts and exhausted through a single shaft by exhaust fans located on the surface. The fresh air supply is divided into four separate streams.

The air drawn down the Air Intake Shaft and the Salt Handling Shaft is split into three separate air streams serving the construction, north area and waste disposal areas. The air drawn down the Waste Shaft serves the Waste Shaft station operation and is exhausted directly to the Exhaust Shaft station, where it joins the exhaust streams of the other three areas. The combined exhaust streams are drawn up the Exhaust Shaft, and discharged directly to the atmosphere under normal operation or via the HEPA filtration system under certain off-normal conditions.

Standby HEPA filtration, also located on the surface, is engaged upon detection of radioactive particulates in the waste disposal exhaust stream.

### 6.3.1 The Normal Mode (Exhaust Filtration Bypassed)

The Normal Mode of ventilation is with the exhaust filtration system bypassed. Five different levels of Normal Mode ventilation can be established to provide five different air flow quantities. These five levels of air flow are achieved by the use of the various exhaust fans as follows:

- **Normal Ventilation**: Two of three main exhaust fans operating to provide 425,000 standard cubic feet per minute (scfm) unfiltered.
- **Alternative Ventilation**: Any one of the three main exhaust fans operating to provide 260,000scfm unfiltered.
- **Reduced Ventilation**: Any two of three filtration fans operating as ventilation fans to provide 120,000scfm unfiltered.
- **Minimum Ventilation**: Any one of three filtration fans operating as a ventilation fan to provide 60,000scfm unfiltered.
- **Maintenance Ventilation**: Any one or two of the three main exhaust fans operating in parallel with one or two of the filtration fans to provide approximately 260,000scfm to 425,000scfm.

### 6.3.2 Filtration Mode

The filtration mode of ventilation is designed to confine airborne radiological contamination released by a breached waste container in the underground, minimizing any release to the environment. Filtration shall be automatically initiated by detection of radioactive airborne contaminants above the set point. A single 860 Series fan provides up to 60,000scfm in filtration mode exhausted through the HEPA bank.

### 6.3.3 Dynamic Pressure Effects

The underground ventilation system is basically a steady state system. When it becomes necessary to make a change in operating mode there are dynamic pressure changes which must be considered. These are primarily only in ventilation, such as a shift to filtration that may cause temporary localized pulses. The magnitude and location of these may be affected by the proximity of the shafts.

On February 5, 2014, the ventilation was in the maintenance mode until a fire was reported in the underground at 1050. When the FSM received notice of smoke in the underground in unexpected locations, he made the decision to switch ventilation to the Filtration Mode at 1058 in an attempt to control and slow the spread of smoke throughout the underground.

**Analysis**

There was a ventilation change made eight minutes into the reported vehicle fire in the underground areas of WIPP. When the ventilation was changed to filtration mode, the ventilation in the mine was reduced from 260,000 scfm to approximately 60,000 scfm. The reduction in ventilation did slow the distribution of the smoke, but had the potential of causing the workers to be exposed to heavier smoke and higher levels of carbon monoxide. This is based

on studies from mining experts when mine fires and explosions occur. This unanalyzed change to filtration mode resulted in a change in air flow in the underground altering the conditions in the primary and secondary evacuation routes. The switch to filtration was unannounced and confused workers as to their proper egress routes.

Early understanding of the exact location of the fire may have enabled decisions on ventilation door closure that would have minimized smoke flow into evacuation routes. The ability to change ventilation configuration remotely to control smoke was hampered by chained doors and a regulator in need of repair, i.e., 707 bulkhead regulator.

**CON 2**: NWP management allows expert-based, rather than a process/systems-based approach to decision making, e.g., shift to filtration during a fire, sheltering decisions, etc.

**JON 3**: NWP needs to evaluate and apply a process/systems based approach for decision making relative to credible emergencies in the U/G, including formalizing response actions, e.g., decision to change to filtration mode during an ongoing evacuation.

## 6.4 Underground Communications and Emergency Notification Systems Description

The Board reviewed the Underground Communications Systems that are in use at the WIPP and were used during the U/G fire of February 5, 2014. The following is a description of the systems in use.

The Central Monitoring System (CMS) is a supervisory control and data acquisition (SCADA) system consisting of a mix of functional units communicating on a redundant network throughout the facility on the surface and in the underground. The network is made up of optical fiber and the associated fiber distribution units, switches, etc. The functional units are LPUs, operator, server PCs, printers and uninterruptible power supplies.

The CMS is used for real-time site data acquisition, display, storage, alarming and for the control of site components. The CMS monitors process, environmental, electrical, mechanical, radiation, and fire protection systems and provides manual and automatic control of underground ventilation, backup power, underground evacuation alarm automatic shift to filtration, and electrical distribution.

The CMR, located on the second floor of the Support Building, is the central location for monitoring site data and conditions. It is the location of the primary man-machine interface with the CMS, Remote Fire Alarm Reporting (RFAR) station, a satellite weather service and a commercial television weather station. The operator is in voice contact with the on-site and off-site activities via the dial phones, mine pager phones, public address and intercom system and two-way radio. The master control console for public address and evacuation alarm control is located in the CMR. Space, phones, and furniture are provided in the CMR for the activities of the Operations Assistance Team during emergency conditions.

The Dial Phones system is a private automatic telephone exchange for on-site and off-site telephone communications. Dial phones and other terminal devices are located throughout the site. The telephone switchgear, backup batteries and battery charger are located in the telephone hut (Building 468) near the Support Building. Telephone communications are carried off of the site by cable and a microwave system that are owned and operated by the telephone company.

Mine pager phones is a network of independent, interconnected, self-contained, battery-powered paging phones used for two-way emergency and routine communication between the underground and the surface. The mine phones are interconnected on a two wire system. Each phone includes a speaker for paging and a handset for initiating pages and for normal phone communication between one or more other mine pager phones on the system. The speaker signal and the handset signal are electronically amplified at each phone.

Plant PA and Alarm Systems includes the site-wide public address installations and a separate and additional underground evacuation alarm system (strobe lights). The public address system master control console is located in the CMR. Submaster paging stations are located in the support building, Waste Handling Building, water pump house, guard and security building, salt handling hoist house and head frame, exhaust filter building, safety and emergency services facility, engineering building, training building, warehouse/shops building, and underground.

The Hoist Radio system is comprised of a wireless, medium frequency FM radio system that provides two-way voice communication between the hoist control room and the shaft conveyance (cage) in the waste-handling and salt-handling shafts. Programmable logic controller and radio modems provide for control of the movement of the waste-handling shaft hoist from the cage for special activities such as shaft inspection and maintenance. The voice radio system uses the hoist rope as a signal path (antenna), and the radio modems use antennae mounted on the cage and at the hoist tower on the surface.

The WIPPnet wide area network provides inter-connectivity between the WIPP, the underground facilities, and in-town buildings. Fiber-optic cable provides connectivity between buildings and the Underground areas at the WIPP site. Microwave and fiber links established through contracts with the local telephone provider provide connectivity between the WIPP site and the in-town network elements.

The EOC is the designated, centralized location from which the site emergency response organization evaluates, coordinates and manages response activities and communicates with DOE and other federal, state, and local organizations. The EOC is located on the site in the safety and emergency services facility. It contains communication devices that are a part of the Dial Phones, Plant PA and Alarm Systems, Mine Pager Phones, and Radio and other systems.

The Board has reviewed documents and statements from the workers that specifically stated that they could not hear the yelps or see the strobe lights, and the messages on the pagers were muffled and could not be understood.

**Analysis**

The procedure to begin evacuation of the underground requires the CMRO to turn on the strobe lights and activate the yelp alarm. The yelp alarm was only activated for about three seconds

instead of the procedurally required five seconds. Within a few minutes of the yelp alarm, the CMRO was notified by one of the workers that the strobe lights were not activated. The CMRO immediately activated the strobe lights. The strobe lights are a critical piece of the communication system in alerting the workers underground of an evacuation. Due to the heavy equipment operations and other activities, the audible alarm could not be heard by everyone underground. Most workers rely on the strobe lights for notification.

The FSM and the CMRO did not fully follow the procedures for response to the fire in the underground. This can be attributed to the complexity of the alarm and communication system, lack of effective drills and training, and additional burdens placed on the FSM due to the lack of a structured Incident Command System. Unreasonable expectations are placed on the FSM and CMRO in an emergency situation. Critical elements of the system should be evaluated and automated.

**CON 16**: There are elements of the CONOPS program that demonstrate a lack of rigor and discipline commensurate with operation of a Hazard Category 2 Facility.

**JON 25**: NWP and CBFO need to evaluate and correct weaknesses in the CONOPS program and its implementation, particularly with regard to flow-down of requirements from upper-tier documents, procedure content and compliance, and expert-based decision making.

**CON 2**: NWP management allows expert-based, rather than a process/systems-based approach to decision making, e.g., shift to filtration during a fire, sheltering decisions, etc.

**JON 3**: NWP needs to evaluate and apply a process/systems based approach for decision making relative to credible emergencies in the U/G, including formalizing response actions, e.g., decision to change to filtration mode during an ongoing evacuation.

**CON 1**: The FSM and Central Monitoring Room Operator (CMRO) did not fully follow the procedures for response to a fire in the U/G. This can be attributed to the complexity of the alarm and communication system, lack of effective drills and training, and additional burdens placed on the FSM due to the lack of a structured Incident Command System (ICS).

**JON 1:** NWP needs to evaluate and correct deficiencies regarding the controls for communicating emergencies to the underground, including the configuration and adequacy of equipment (alarms, strobes, and public address).

**JON 2**: NWP needs to evaluate the procedures and capabilities of the FSM and CMRO in managing a broad range of emergency response events through a comprehensive drill and requalification program.

# 7.0 NWP Contractor Assurance System

The NWP Contractor Assurance System (CAS) is described in the CBFO approved Quality Assurance Program Description (QAPD), Section 1.1.9. This section captures the criteria specified in the Contracts Requirements Document of DOE Order 226.1B, *Implementation of Department of Energy Oversight Policy.* The CAS commits to ensuring that work performance meets the applicable requirements for environment, safety, and health; integrated safety management; safeguards and security; and emergency management. The CAS states that it is designed to identify deficiencies and opportunities for improvement, report deficiencies to responsible managers, complete corrective actions, and share in lessons learned.

The Contracts Requirements Document of DOE O 226.1B requires the contractor to submit to DOE for approval a CAS Description Document. The contractor, NWP, utilizes the Quality Assurance Program Description (QAPD) to meet this requirement. The QAPD does not refer to other procedures or processes on how the CAS is executed.

The Board reviewed additional resources and found that NWP has numerous policies, procedures and tools for conducting supervision and oversight of work. The Board reviewed several mechanisms on the WIPP Intranet such as lessons learned (many types and databases), trending reports, surveillance plans, and environment, safety and health tools, for example: automated job hazards analysis, radcon, health services, industrial safety, and industrial hygiene databases. NWP also implements other oversight and management processes like quality assurance, CONOPS, WIPP forms/logs, root cause analysis, and environmental management systems.

The Board reviewed the NWP CAS implementation and found the following issues that have not been corrected:

- Multiple external reviews have identified deficiencies in Work Planning & Control, Emergency Management, Issues Management, and Fire Protection.
- Post-drill emergency exercises did not identify deficiencies in the emergency response program, e.g., functionality of egress strobe lights, reflectors, PA system, donning SRs and SCSRs.
- The Emergency Program triennial program assessment was not performed, and it is indeterminate when the last assessment was conducted.
- Combustible material was allowed to build up on non-waste haul vehicles, and in addition, combustible material was allowed to build up in some areas of the Underground.
- Thirty-three emergency lights in the waste handling building have been inoperable for as long as two years.
- Twelve of 40 mine phones tested were found to be non-functional in a spot check by the Board.
- Pre-operational underground vehicle check list did not include performance criteria from the owner's manual.

- There were over 10 red tags related to critical safety equipment posted in the CMR. Some were seven months old. Critical safety equipment includes, but is not limited to, ventilation fans, fire suppression systems, bulkhead doors, and continuous air monitors.
- Lessons Learned from previous underground vehicle fires were not applied.
- Salt haul trucks are designed and built to use fire resistant hydraulic fluid, but it was not used in non-waste haul trucks.
- Surveillances and oversight are more focused on waste-handling and certification activities and less on maintenance activities and the safe operation of the mine.

## 7.1 NWP Supervision and Oversight of Work

NWP has numerous policies, procedures and tools for conducting supervision and oversight of work. The Board reviewed several mechanisms on the WIPP Intranet such as lessons learned (many types and databases), trending reports, surveillance plans, and environment, safety and health tools, for example: automated job hazards analysis, radcon, health services, industrial safety, and industrial hygiene databases. NWP also implements other oversight and management processes like quality assurance, CONOPS, WIPP forms/logs, root cause analysis, and environmental management systems.

An area that the Board specifically reviewed was the Management Assessment Program for NWP. The data that were analyzed included an interview with the Performance Assurance Manager as well as information provided on the WIPP intranet. This manager's duties include occurrence reporting processing system, and there is a Facility Management Designee (FMD) who fulfills and has ownership of this program. He also has the Directive Management Processes where he would ensure and track the implementation of the DOE Directives within the NWP contract. The FMD told the Board that he has the Lessons Learned Program, the Root Cause Analysis Process, and he is the Chairman of the Senior Managers Corrective Action Review Board. The Price-Anderson Amendments Act (PAAA) Coordinator also reports to the FMD and has combined responsibility for Security, Nuclear Safety, and Worker Safety. The WIPP does not protect classified material, it protects nuclear material. Each of the group managers performs the assessments for his/her own group.

The Board has reviewed Attachment 1 of the Management Assessment performance indicator chart. The quality of Management Assessment reports and compliance to program requirements continues to improve per the reviewed 2013 report. The FMD duties seem to be excessive and are performed with little assistance.

Results from the Management Assessment: Based on the review of ten NWP management assessment reports, and on the results of the independent audit, the management assessment team concluded that the weaknesses identified in the UCOR ISMS I\II Review (Finding QA-P2-06) are not prevalent in the implementation of the NWP Management Assessment Program.

Overall, NWP expends considerable resources performing oversight activities, most of which are focused on waste management and quality assurance activities to ensure permit requirements are met.

**Analysis**

The Board determined that the progress toward effectively implementing Work Planning & Control, Emergency Management, Issues Management, and Fire Protection programs is inadequate. NWP has not fully developed a CAS that provides assurance to both DOE and NWP that work is performed compliantly, risks are identified and managed, and control systems are effective and efficient.

Overall, NWP expends considerable resources performing oversight activities, most of which are focused on waste management and quality assurance activities to ensure permit requirements are met.

**CON 14**: NWP has not fully developed an integrated contractor assurance system that provides assurance that work is performed compliantly, risks are identified, and control systems are effective and efficient.

**JON 23**: NWP needs to develop and implement a fully integrated contractor assurance system that provides DOE and NWP confidence that work is performed compliantly, risks are identified, and control systems are effective and efficient.

# 8.0 DOE Programs and Oversight

## 8.1 CBFO Facts

The Carlsbad Field Office (CBFO) provides primary oversight to the site contractor Nuclear Waste Partnership (NWP) and its subcontractors. Day-to-day oversight of field activities at the site is mostly completed by the CBFO staff from the Office of Site Operations and the Office of Environment, Safety, and Health within the CBFO. The CBFO manager has implemented a practice to be at the site at least twice a week.

CBFO oversight staff members include a diverse set of talents and backgrounds including: facility representatives, systems engineering, mine operations, waste operations, work control, quality assurance, electrical safety, environmental protection, regulatory specialist, RCRA, compliance, emergency management, fire protection, health physics, and safety.

CBFO has several policies and procedures that address oversight activities such as QA audits, surveillances, and other project verifications. CBFO is required to implement an oversight program in accordance with DOE O 226.1B. CBFO also implements a Technical Qualification Program (TQP) in accordance with DOE O 426.1.

Per the CBFO Integrated Safety Management System Description, DOE/CBFO 09-3442, Rev. 3, *Introduction*:

> *"The CBFO mission is to provide safe, compliant, and efficient characterization, transportation, and disposal of defense transuranic (TRU) waste. CBFO is committed to fulfilling its mission in a manner that affords protection of the public, our Federal, contractor, and subcontractor worker, and the environment. CBFO is dedicated to performing its mission in compliance with the statues enacted by Congress for the protection of workers, the public, and the environment, and for exercising good stewardship of public property. This protection is put into operation at all levels (site, facility, task, and activity) by requiring and routinely verifying that work is conducted following the five ISM Core Functions in a manner consistent with the seven ISM Guiding Principles established in DOE P 450.4."*

The Board interviewed several of the CBFO management and oversight staff and reviewed numerous documents during the course of this investigation. Periodically, CBFO oversight functions are supplemented by DOE-HQ, DNFSB, DOE-EMCBC, MSHA, and other outside entities to ensure safe and compliant operations at the facility.

The Waste Isolation Pilot Plant Land Withdrawal Act, Public Law 102-579, and a Memorandum of Understanding (MOU) between the U.S. Department of Energy and the U.S. Department of Labor (dated July 1987) state, in part, that MSHA will shall inspect WIPP not less than four times each year and in the same manner as it evaluates mine sites under the Federal Mine Safety and Health Act of 1977, and shall provide the results of its inspections to DOE so DOE can implement its policy of compliance to MSHA standard (as though WIPP was a commercial mine) by taking the necessary actions with the DOE contractors and to assure the prompt and

effective correction of any deficiencies and to otherwise ensure general compliance with MSHA's mining health and safety requirements.

CBFO and EMCBC have signed a Service Level Agreement (SLA) that describes support functions to be provided by EMCBC in order for CBFO to be able to focus its resources on project and technical management, and oversight of CBFO contractors. The SLA describes EMCBC functions such as support in the areas of regulatory compliance, safety management systems, quality assurance, lessons learned, contractor assurance, technical support, and DOE oversight assistance. The SLA also states the EMCBC can provide preparation, review and issuance of program procedures and plans, as required to support the mission and conduct/support audits and surveillances per DOE management guidance.

DOE Headquarters provides support to WIPP in the form of policies, DOE orders, resources, mission support, emergency management, and independent oversight. DOE HQ does not currently provide resources to WIPP that address the unique challenge of operating a Hazard Category 2 facility in a mine.

**Analysis**

The Board reviewed the CBFO Integrated Evaluation Plans from FY11 to the present to assess the completion status of planned assessments. While several of the scheduled assessments were completed and documented, many of the scheduled evaluations logged within the Integrated Evaluation Plans. Examples included scheduled senior management walkthroughs, Safety System Oversight (SSO) for ventilation, nuclear safety management program review, Office of Site Operations (OSO) management assessment, vital safety systems (VSS) walk down of CAMS systems, Technical Qualification Program (TQP) assessments, Maintenance procedure assessment, FHA/BNA assessment, etc., were completed as listed on the Plan).

In addition, from interviews with several CBFO staff members, there is a strong perception that contractor and mid-level CBFO management do not welcome negative findings or observations and that CBFO staff have to individually follow up on corrective actions from NWP (rather than getting timely responses in accordance with site corrective action processes) in order to ensure effective actions have been taken. It was not apparent that follow-up is pursued in all cases by CBFO staff. Several CBFO staff members indicated that they can convey issues verbally to the contractor with mixed results for correction; however, there is not an effective mechanism to convey documented issues to the contractor. In addition, from review of the recent Safety Conscious Work Environment employee survey, 59 percent of the CBFO staff members that completed the survey answered "somewhat" to "yes" on the question of the existence of a chilled work environment.

CBFO staff members have been required to use the Office of Quality Assurance corrective action report (CAR) system to identify nonconformances. Interviews with several CBFO staff members indicate that this process is cumbersome, administratively burdensome, and many do not use it. In reviewing CAR submittals since the beginning of FY2012, the Board found that only 15 CARS have been generated by site staff outside of the CBFO QA group. Only one CAR has been generated by a facility representative in the last year.

The Facility Representative program has been reviewed several times over the last few years. Deficiencies have been identified related to staffing not meeting the staffing analysis, procedures that are incomplete and not used, no structured surveillance/oversight program, and no clear mechanism being used to communicate issues to management and the contractor (see Table 3). While CBFO management has brought in supplemental support from HQ and EMCBC to try to correct these issues, the FR program is still not effectively implemented.

Several externally (DOE-HQ, DNFSB, HS, EMCBC, etc.) generated oversight documents that contained findings, observations, and opportunities for improvement for the CBFO and WIPP site were reviewed by the Board. In many cases, no corrective action plans were developed or implemented, corrective action responses were not developed in a timely manner (for example, a year lapsed between the assessment and development of a corrective action plan), or implementation of corrective actions was either incomplete or ineffective. Several of the deficiencies have been identified numerous times. Table 3 includes examples of external oversight reports that were reviewed by the Board.

**Table 3: Reviews of the WIPP Project**

| Date of External Assessment | | External Assessment Title | Areas Evaluated |
|---|---|---|---|
| January 26 – 30, 2009 | EM-43 | Environmental Management Quality Assurance Audit<br>Department Of Energy<br>Carlsbad Field Office<br>Washington TRU Solutions and Central Characterization Project<br>EM-PA-09-013 | Quality Assurance (QA) audit of Planning and Control |
| March 31, 2009 | EM-64<br>(EM-43) | Environmental Management Quality Assurance Program Audit of the Waste Isolation Pilot Plant<br>Transmittal Letter | Flowdown of requirements; adequacy of CBFO oversight of the QA program; appropriateness of the interface controls; adequacy of purchase items; and adequacy of identifying conditions adverse to quality. |
| March 9-12, 2010 | EM-22<br>(EM-42) | Waste Isolation Pilot Plant<br>Washington TRU Solutions, LLC<br>EM-22 Office of Safety Operations Assurance Assessment Report | Ongoing and regular evaluation of the effectiveness of the WIPP operations.<br>Evaluated CONOPS, Radiological Protection, Work Planning and Control Programs, and CBFO oversight. |
| February 15-17, 2011 | EM-22<br>(EM-42) | EM-22 Office of Safety Operations Assurance<br>Waste Isolation Pilot Plant Review | Evaluate Washington TRU CONOPs, Work Planning and Control and Contractor Assurance System processes.<br>Follow-up to March 2010 EM-22 assessment. |

| Date of External Assessment | | External Assessment Title | Areas Evaluated |
|---|---|---|---|
| June 24, 2011 | DNFSB | Forwarding the Staff Issue report for a staff review conducted January 25-26, 2011, on the fire protection program at WIPP, including both above-ground and underground operations. | Identified issues with the Fire Hazard Analysis, contractor's fire protection program, CBFO oversight, WIPP fire brigade, baseline needs assessment, and CBFO's emergency management program. |
| September 7, 2011 | HSS | Office of Enforcement and Oversight conducted an orientation visit to the DOE Carlsbad Field Office (CBFO) and the nuclear facility at the Waste Isolation Pilot Plant (WIPP). | The purpose of the visit was to discuss the nuclear safety oversight strategy, describe the site lead program, increase HSS personnel's operational awareness of the site's activities, and identify specific activities that HSS can perform to carry out its independent oversight and mission support responsibilities. |
| May 7-10, 2012 | MSHA | Mine Safety and Health Administration (MSHA) inspection of surface and underground safety systems | 9 underground Compliance Assistance Visit (CAV) notices and 9 surface CAV notices. |
| July 23-26, 2012 | EM-42 | EM-22 Office of Safety Operations Assurance<br>Waste Isolation Pilot Plant<br>Maintenance Management Review | Evaluate the Washington TRU Solutions Maintenance Management Program and the CBFO oversight of this program.<br>Prompted by June 27, 2012 letter from DNFSB to Senior Advisor for EM detailing safety issues with the site. |

| Date of External Assessment | | External Assessment Title | Areas Evaluated |
|---|---|---|---|
| October 5, 2012 | EMCBC | The assessment was completed at the request of the CBFO Manager, and was covered over a period of time of August 6-9, 2012. | The review was conducted on safety programs and oversight implementation in response to a previous organizational assessment and due to concerns reported through the EMCBC Employee Concern Program. |
| November 12-15, 2012 | EM-42 | EM-42 Office of Operational Safety<br>Waste Isolation Pilot Plant Maintenance Management Assist Visit | Evaluate the status of commitments made by EM Senior Advisor for EM in September 2012 in response to the DNFSB June 24, 2012, letter detailing actions taken and planned to correct to issues with the WIPP maintenance management program. |
| November 29, 2012 | HSS | Independent Oversight review of Site Preparedness for Severe Natural Phenomena Events at the Waste Isolation Pilot Plant – November 2012 | Office of Enforcement and Oversight independent oversight review of the WIPP emergency management program during June 5 –July 12, 2012. The HSS Office of Safety and Emergency Management Evaluations performed this review to evaluate the processes for identifying emergency response capabilities and maintaining them in a state of readiness in case of a severe NPE. |

| Date of External Assessment | | External Assessment Title | Areas Evaluated |
|---|---|---|---|
| January 14-18, 2013 | HS-12 (VPP) | DOE-HSS evaluation of security Walls Voluntary Protection Program (VPP) | Security Walls (security contractor under Washington TRU Solutions (WTS)) had received the Star Level under VPP but gave it up when they became a part of NWP. NWP has a transition plan in place as part of the new contract and received a legacy award in August 2013 for the transition plan. They will need to meet additional criteria including completing the ISMS implementation verification and validation reviews. |
| April 2013 | EM-43 | Follow-Up Assessment of QAP Implementation at the Department of Energy Environmental Management Carlsbad Field Office in Carlsbad, New Mexico, EM-PA-12-14, January 28-31, 2013 | Follow up assessment of implementation of the QAP. |
| April 2013 | HSS | Report documenting 2 onsite reviews: first on June 25-28, 2012, and a follow-up visit on January 22-24, 2013. | Objectives of the Independent Oversight review were to evaluate selected portions of 1) CBFO's oversight of the contractor's effectiveness review documentation; and 2) CBFO's performance of the annual ISMS declaration review of the contractor's work planning and control element. |

| Date of External Assessment | | External Assessment Title | Areas Evaluated |
|---|---|---|---|
| June 2013 | EM-42 | WIPP CBFO Oversight And Management Assist Visit | The team found continued immaturity in the CBFO oversight and issues management processes which resulted in a burdensome process for FR issues to be transmitted to the CBFO management and contractor. |
| June 11-13, 2013 | EM-42 | EM-42 Office of Operational Safety<br>Waste Isolation Pilot Plant Carlsbad Field Office Oversight and Maintenance Management Assist visit | Provide assistance to the DOE Carlsbad Office in improving its oversight of NWP operations at WIPP. |
| June 27, 2012 | DNFSB | Forwarding the Staff Issue Report for an on-site review conducted during the week of march 5, 2012, on the WIPP maintenance program. | Deficiencies were identified by the staff with respect to quality of and compliance with maintenance work control documents, post-maintenance testing, pre-job reviews, annual system walk downs, maintenance resources, placekeeping, and DOE oversight. |
| July 2013 | EM-42 | Triennial Assessment of the CBFO Facility Representative Program | EM-42 staff was requested by the CBFO to perform this assessment in accordance with DOE-STD-1063. |
| August 19-29, 2013 | EM-44 | Verification of WIPP Assessment for HS-45 and EM-44 Corrective Actions | Review of corrective actions identified by HSS-45 and EM-44 regarding the implementation of an integrated and comprehensive Emergency Management Program |

| Date of External Assessment | | External Assessment Title | Areas Evaluated |
|---|---|---|---|
| August 22, 2013 | DNFSB | DNFSB Staff visit on WIPP Status | Areas of discussion included work planning and control, fire protection, plans and concepts for WIPP's future, DOE-CBFO contractor oversight program, and underground and above-ground tours. |
| January 28-30, 2014 | MSHA | MSHA inspection of surface and underground safety systems | CAV notices have been transmitted to CBFO but have not yet been processed into corrective actions by CBFO. |

Per the MOU and Land Withdrawal Act, MSHA is required to provide the site with safety inspections no less than four times per year. Records indicate that MSHA has only performed inspections two times over the last three years. This is a missed opportunity to identify mine safety issues from experienced inspectors. CBFO does not have equivalent resources to perform this function, nor have they identified the lack of MSHA oversight support as an issue that needs attention.

At the request of the CBFO manager, EMCBC provided a Line Management Oversight Review in October 2012 that identified several weaknesses in oversight programs and implementation. Subsequent to the issuance of this report, there has been inadequate follow up to ensure that CBFO was provided the necessary technical and oversight support functions as described in the SLA.

Overall, CBFO needs to establish and implement an effective line management oversight program and processes that meet the requirements of DOE O 226.1B and hold personnel accountable for implementing those program and processes.

DOE HQ needs to ensure that adequate resources are available for mission support (e.g. specialized expertise to support WIPP's unique work scope, and resources to ensure safe mine operations) and that projects are held accountable for effective and timely corrective actions to issues identified during independent oversight activities.

**CON 15**: CBFO failed to adequately establish and implement line management oversight programs and processes to meet the requirements of DOE O 226.1B and hold personnel accountable for implementing those programs and processes.

**JON 24**: CBFO needs to establish and implement an effective line management oversight program and processes that meet the requirements of DOE O 226.1B and hold personnel accountable for implementing those programs and processes.

**CON 16**: CBFO management does not have adequate communication processes to ensure awareness of issues that warrant attention from all levels of the DOE staff.

**JON 25**: CBFO needs to accelerate the implementation of a mechanism for all levels of CBFO staff to document, communicate, track, and close issues both internally and with NWP.

**JON 26**: The CBFO Site Manager needs to institutionalize and communicate expectations for the identification, documentation, reporting, and correction of issues.

**CON 17**: DOE HQ failed to ensure that CBFO was held accountable for correcting repeatedly identified issues involving fire protection, maintenance, emergency management, work planning and control, and oversight.

**JON 27**: DOE HQ needs to ensure that repeatedly identified issues related to safety management programs (SMPs) are confirmed closed and validated by the local DOE office.

This process should be considered for application across the DOE complex and include tracking, closure, actions to measure the effectiveness of closure (line management accountability), and trending to identify precursors and lessons learned.

**JON 28**: DOE HQ should enhance its required oversight to ensure site implementation of the emergency management policy and requirements are consistent and effective. Emphasis should be placed on ensuring ICSs are functioning properly and integrated exercises are conducted where personnel are evacuated.

**CON 18**: DOE HQ failed to ensure CBFO was provided with qualified technical resources to oversee operation of a Hazard Category 2 Facility in a mine.

**JON 29**: DOE HQ needs to develop and implement a process for ensuring that technical expertise is available to provide support in the unique area of ground control, underground construction, and mine safety and equipment.

**JON 30**: DOE HQ needs to assist CBFO with leveraging expertise from Mine Safety and Health Administration (MSHA), in accordance with the DOE/MSHA MOU, in areas of ground control, underground construction, and mine safety where DOE does not have the expertise.

**JON 31**: DOE HQ needs to re-evaluate resources (i.e. funding, staffing, infrastructure, etc.) applied to the WIPP project to ensure safe operations of a Hazard Category 2 facility.

**CON 19**: The Office of Environmental Management Consolidated Business Center (EMCBC) and CBFO failed to ensure support services as described in the Service Level Agreement were provided.

**JON 32**: EMCBC and CBFO need to develop and implement clear expectations and a schedule for EMCBC to provide support in the areas of regulatory compliance, safety management systems, preparation of program procedures and plans, quality assurance, lessons learned, contractor assurance, technical support, DOE oversight assistance, etc.

# 9.0 Safety Programs

## 9.1 Integrated Safety Management Systems

NWP is required to implement a Safety Management System in accordance with 48 CFR 970.5223-1, *Integration of Environment, Safety, and Health into Work Planning and Execution*. The requirement states that in performing work, the contractor shall perform work safely, in a manner that ensures adequate protection for employees, the public, and the environment, and shall be accountable for the safe performance of work. The contractor shall ensure that management of Environment, Safety and Health (ES&H) functions and activities becomes an integral but visible part of the contractor's work planning and execution processes. The five core safety management functions provide the necessary structure for any work activity, including emergency management, which could potentially affect the public, the workers, and the environment.

NWP has not had its (Integrated Safety Management System) ISMS program verified through the DOE ISMS verification process. The ISMS verification was scheduled for May 2013, and later rescheduled for September 2013. The NWP ISMS verification is currently scheduled for May 2014.

NWP and CBFO completed a joint ISMS and QA Declaration for FY12. This declaration concluded that ISMS and QA programs have been implemented and are effective at ensuring safety and quality performance. This declaration was based on multiple external and internal reviews. One joint external review conducted by the DOE Office of Health, Safety and Security (HSS) and CBFO identified 82 issues with NWP's implementation of Work Planning and Control program. This external review also identified a finding in which CBFO did not follow its internal process for documenting findings. NWP and CBFO had not yet completed their FY13 annual ISMS and QA declaration. However, NWP reached back to URS corporate to conduct an assessment of the Work Planning and Control process that concluded improvements in the Work Planning and Control program.

**Analysis**

The Board highlighted the following deficiencies with each of the five core functions (CF) and its applicable guiding principle (GP).

| **Define the Scope of Work (CF-1)** |
|---|
| Line Management is Responsible for Safety (GP-1) |
| Competence Commensurate with Responsibilities (GP-3) |
| Balanced Priorities (GP-4) |
| Identification of Safety Standards and Requirements (GP-5) |

- NWP and CBFO did not effectively establish a work environment where the requirements for nuclear safety, mine safety, and occupational safety are integrated and understood by their employees.
- NWP and CBFO did not ensure that emergency training and drills were performed such that employees were able to respond and evacuate the U/G during an actual emergency condition.

**Identify and Analyze the Hazards Associated with the Work (CF-2)**

Identification of Safety Standards and Requirements (GP-5)

Hazard controls tailored to work performed (GP-6)

- NWP did not implement a pre-operational vehicle use checklist process in accordance with the vehicle manufacturer's instructions.
- The Fire Hazard Analysis did not consider the impact of a vehicle fire near the Air Intake Station.
- NWP failed to recognize the consequences of not maintaining U/G vehicles in accordance with manufacturer's instructions.
- NWP did not fully analyze and develop response plans to various emergency scenarios.

**Develop and Implement Hazard Controls (CF-3)**

Identification of Safety Standards and Requirements (GP-5)

Hazard controls tailored to work performed (GP-6)

Operations authorized (GP-7)

- The emergency response procedures did not clearly define points when U/G ventilation should be secured and/or changed, egress methods for conditions when multiple people are in the U/G, and when to activate the EOC.
- NWP did not implement its housekeeping program such that egress is not impeded and combustible loading is not exceeded.
- General employee training and the U/G fire response procedure are inconsistent in regard to responding to an incipient-stage fire.

**Perform Work within Controls (CF-4)**

Clear Roles and Responsibilities (GP-2)

Competence commensurate with responsibilities (GP-3)

Operations authorized (GP-7)

- The management systems supporting the decision to change ventilation during an emergency condition did not require the FSM to consult with the subject matter expert (SME) and U/G personnel.
- U/G vehicle pre-operational use checklists were not performed in accordance with the manufacturer's instructions, including the verification of vehicle performance criteria, e.g., oil pressure.
- U/G personnel were unable to don SRs and/or SCSRs.
- Haul truck operator did not notify the CMR of the fire, after failure of the portable fire extinguisher and the vehicle fire suppression system.

**Feedback and Improvement (CF-5)**

Line Management is Responsible for Safety (GP-1)

- Multiple opportunities were missed to mitigate the hazards and risks associated with the pre-operational condition of U/G vehicles.
- Multiple opportunities were missed to apply Lessons Learned from other events when U/G vehicles caught on fire.
- NWP did not adequately evaluate the effectiveness of training in donning and use of SR and/or SCSR in the U/G.
- NWP did not fully develop a Contractor Assurance System where both DOE and NWP are assured that work is performed compliantly, risks are identified and managed, and control systems are effective and efficient.
- CBFO has not fully established an oversight program that effectively evaluates the health and effectiveness of CBFO and NWP management systems, and fosters an environment where issues are raised, track and trended, and effectively resolved.

## 9.2 Conduct of Operations Implementation

Operations of the WIPP are described in the WP 04-CO.01, *WIPP Conduct of Operations* procedure series. The series includes procedures for Shift Routines and Operating Practices, Control Area Activities, Communications, Control of On-Shift Training, Notifications, Control of Equipment and System Status, and Operations Procedures. In accordance with DOE Order 422.1, *Conduct of Operations*, NWP has a CBFO-approved CONOPS matrix.

The Board reviewed the CONOPS program and identified the following:

- Maintenance procedure PM074080, EMCO Haul Truck 74-U-006A/B, does not refer to the CHAMPS Preventative Maintenance process, nor include performance requirements from manufacturer's instructions. While "Various O&M Manuals" are listed as a reference in the procedure, there are no steps in the procedure that direct the user to refer to the manufacturer's instructions and validate performance criteria.

- Operator's Checklists are not completely filled out. On several occasions, the initial and/or final meter reading was not filled out, and the salt haul truck is not marked as safe to use.
- The emergency response procedures did not clearly identify points when U/G ventilation should be secured and/or changed, egress methods for conditions when multiple people are in the U/G, or when to activate the EOC.
- The BNA requirement for use of the manual onboard FSS before use of a portable fire extinguisher was not included in the U/G fire response procedure.
- The U/G fire response procedure required the CMRO to direct U/G Services to respond and evaluate fires after a decision was already made to evacuate the mine.
- As identified in training and written in procedures, the haul truck operator did not notify the CMR of the fire after the portable fire extinguisher and manual FSS failed. Instead, the Operator contacted the maintenance department.
- Although required by the evacuation procedure, the CMR did not sound the evacuation alarm for a full five seconds and illuminate the emergency strobe lights.
- Many U/G personnel were unable to don SRs and/or SCSRs.
- Critical safety equipment had red tags in which NWP employees via interviews did not fully understand the status of the impaired safety-related equipment. Safety equipment included fans, FSS, and CAM.
- The U/G haul truck operator did not receive hands-on training on the use of portable fire extinguishers.

**Analysis**

The elements of the NWP CONOPS program reviewed by the Board indicate weaknesses in implementation. NWP has not developed procedures and processes that ensure:

- U/G vehicles are maintained in accordance with the manufacturer's instructions.
- Emergency drill U/G evacuations demonstrated proficiency in donning of SRs and SCSRs, activating alarms and lights, making DOE notifications, and activating the EOC.
- Emergency condition procedures, e.g., U/G Fire, Mine Evacuation, could be executed without expert-based decision making.
- FSM fully understood impacts of changing ventilations modes while personnel are in the U/G during an emergency condition.
- BNA and FHA requirements are flowed in implementing documents.

The Board determined that NWP approached CONOPS from different perspectives, not fully understanding that the entire WIPP facility is a Hazard Category 2 facility. Interviews with workers indicated that the terminology "operations" primarily referred to those daily activities, resources, management, and communication needed to support TRU waste storage operations. This disconnect has reduced the level of rigor applied to operations that are not related to TRU waste handling.

**CON 20**: There are elements of the CONOPS program that demonstrate a lack of rigor and discipline commensurate with operation of a Hazard Category 2 Facility.

**JON 33**: NWP and CBFO need to evaluate and correct weaknesses in the CONOPS program and its implementation, particularly with regard to flow-down of requirements from upper-tier documents, procedure content and compliance, and expert-based decision making.

## 9.3 Human Performance Improvement

The goal of Human Performance Improvement (HPI) is to facilitate the development of a facility structure that recognizes human attributes and develops defenses that proactively manage human error and optimize the performance of individuals, leaders, and the organization. The Department's *Human Performance Improvement Handbook*, Volumes 1 and 2 (DOE-HDBK-1028-2009), describe the HPI tools available for use at DOE sites. The Board did not look at HPI from the perspective of program implementation. The Board evaluated Human Performance to determine if it played a part in this accident. Human error is not a cause of failure alone, but rather the effect or symptom of deeper trouble in the system. A review of Human Performance is a review of an individual's abilities, tasks, and operating environment to determine if the organization supports them for success.

The significance, or severity, of a particular event lies in the consequences suffered by the physical plant or personnel, not the error that initiated the event. The error that causes a serious accident and the error that is one of hundreds with no consequence can be the same error that has historically been overlooked or uncorrected. In most cases, for a significant event to occur, multiple breakdowns in defenses must first occur. Whereas human error may trigger an event, it is the number and extent of flawed defenses that dictate the severity of the event. The existence of many flawed defenses is directly attributable to weaknesses in the organization or management control systems. The Anatomy of an Event Model (Figure 21) illustrates the elements that exist before an event occurs and is a very useful model to guide the analysis of an event from an HPI perspective. The elements analyzed are the flawed defenses that allowed the event to occur or did not mitigate the consequences of the event; the error precursors that existed; the latent organizational conditions that allowed those to be in existence; and finally the vision, beliefs and values of management and workers.

Much of the information provided in this section is based on the analysis of the events, conditions, processes, and barrier information presented in this report.

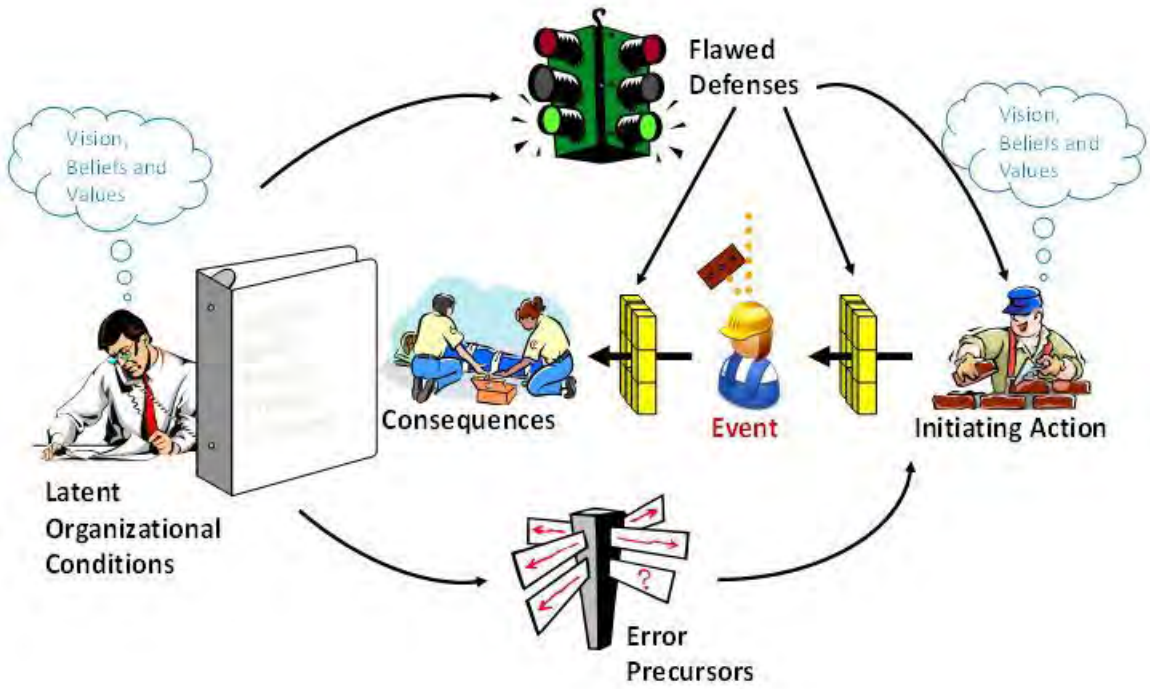



**Figure 27: Anatomy of an Event Model**

### 9.3.1 Error Precursors

Error precursors are unfavorable conditions that increase the probability for error during a specific action and create what are known as error-likely situations. An error-likely situation typically exists when the demands of the task exceed the capabilities of the individual or when work conditions exceed the limitations of human nature. Human nature comprises all mental, emotional, social, physical, and biological characteristics that define human tendencies, abilities, and limitations. For instance, humans tend to perform poorly under high stress and undue time pressure. Error-likely situations such as these are also known as error traps. Error precursors exist in the work place before the error occurs, and thus are manageable. If identified before or during the performance of work, the conditions can be changed or managed to reduce the chance for error(s) leading to an event.

Error precursors (conditions) associated with Human Performance attributes were analyzed by the Board to identify specific conditions that may have provoked error and led to the accident (Figure 28).

### 9.3.2 Human Performance Attributes

**Task Demands**. Specific mental, physical, and team requirements to perform an activity that may either exceed the capabilities or challenge the limitations of human nature of the individual assigned to the task; for example, excessive workload, hurrying, concurrent actions, unclear roles and responsibilities, or vague standards.

**Individual Capabilities**. Unique mental, physical, and emotional abilities of a particular person that fail to match the demands of the specific task; for example, unfamiliarity with the task, unsafe attitudes, level of education, lack of knowledge, unpracticed skills, personality, inexperience, health and fitness, poor communication practices, or low self-esteem.

**Work Environment**. General influences of the workplace, organizational, and cultural conditions that affect individual behavior; for example, distractions, awkward equipment layout, complex tagout procedures, at-risk norms and values, work group attitudes toward various hazards, or work control processes.

**Human Nature**. Generic traits, dispositions, and limitations of being human that may incline individuals to err under unfavorable conditions; for example, habit, short-term memory, fatigue, stress, complacency, or mental shortcuts.

**HUMAN PERFORMANCE ATTRIBUTES**

**Task Demands**. Specific mental, physical, and team requirements to perform an activity that may either exceed the capabilities or challenge the limitations of human nature of the individual assigned to the task; for example, excessive workload, hurrying, concurrent actions, unclear roles and responsibilities, or vague standards.

**Individual Capabilities**. Unique mental, physical, and emotional abilities of a particular person that fail to match the demands of the specific task; for example, unfamiliarity with the task, unsafe attitudes, level of education, lack of knowledge, unpracticed skills, personality, inexperience, health and fitness, poor communication practices, or low self-esteem.

**Work Environment**. General influences of the workplace, organizational, and cultural conditions that affect individual behavior; for example, distractions, awkward equipment layout, complex tagout procedures, at-risk norms and values, work group attitudes toward various hazards, or work control processes.

**Human Nature**. Generic traits, dispositions, and limitations of being human that may incline individuals to err under unfavorable conditions; for example, habit, short-term memory, fatigue, stress, complacency, or mental shortcuts.

**Figure 28: Human Performance Attributes**

### 9.3.3 Error Precursor Analysis

The Board conducted an Error Precursor Analysis based on the information obtained from documents and interviews as documented throughout this report. The results of this analysis are

presented in Table 4. The analysis resulted in the identification of 21 different error precursors on the day of the accident. Four of the identified error precursors existed more than one time that day. The following is a discussion of some of the more predominant error precursors.

### 9.3.4 Human Performance Mode

Human Performance describes three modes in which errors occur. The performance mode in which an error occurs is based on the individual's familiarity with the task being performed. The three modes, progressing from most familiar to the task to the least familiar to the task are: skill based, rules based, and knowledge based. Errors will most likely occur in the knowledge based performance mode.

1. **Donning SRs and SCSRs**. Underground workers were familiar with the use of the SRs and SCSRs. They understood and had been trained in the steps required to don the SRs and SCSRs. During the fire, in some cases, the SRs and SCSRs could not be opened or were not used. Underground workers failed to recognize how changes (e.g. stress, smoke) could complicate donning the SRs and SCSRs. In some instances, the decision was made not to use the SR due to the belief that the individual could reach "good" air quicker by not donning the SR. The Board later determined that SR and SCSR training was not sufficient and that there was no hands-on training that simulates use in likely emergency conditions (i.e., limited visibility due to dark or smoke filled areas). The annual refresher is a video that does not require donning of the SCSR.

2. **Use of Fire Extinguisher**. The Operator did not have adequate training in the use of appropriate fire suppression systems and portable fire extinguishers. The Operator training on the use of the manual suppression system on the salt haul truck was not clear. Workers received video training on use of fire extinguishers; however, they had to rely on assumptions to make a decision on the correct use of a fire extinguisher. No hands-on training had been provided.

3. **Changing Ventilation to Filtration Mode**. FSM did not have all the information necessary to make an informed decision on changing the ventilation mode during the evacuation from the mine. The FSM relied on assumptions and analytical skills to make a decision to reduce smoke from the fire. The unannounced change in ventilation to filtration mode was not in any procedure and quite possibly contributed to higher local concentrations of smoke and carbon monoxide in the drifts. The procedure used in the CMR did not anticipate a full spectrum of potential emergency situations. FSM did not solicit input from other knowledgeable individuals to better understand the conditions or potential impact of the ventilation mode change on the U/G conditions.

4. **Allowing combustible fluid leaks and buildup of combustible "grime" on salt haul truck**. The Operator did not identify any conditions associated with fluid leaks or "grime" buildup on the salt haul truck during pre-use inspections. The frequency of fluid leaks and buildup of "grime" was known by workers. This issue did not get addressed and over time the expectations associated with the condition of the salt haul truck were relaxed to accept these conditions.

**Table 4: Error Precursors**

| TASK DEMANDS (TD) | | | INDIVIDUAL CAPABILITIES (IC) | | |
|---|---|---|---|---|---|
| x[1] | 1 | Time Pressure (In a hurry) | xx | 1 | Unfamiliarity with Task/First time |
| xx | 2 | High Workload (large memory) | xx | 2 | Lack of Knowledge (faulty mental model) |
| x | 3 | Simultaneous, Multiple Tasks | xx | 3 | New Technique not used before |
| | 4 | Repetitive Actions/Monotony | x | 4 | Imprecise Communications |
| x | 5 | Irreversible Acts | xx | 5 | Lack of Proficiency/Inexperience |
| xx | 6 | Interpretation Requirements | | 6 | Indistinct Problem-solving Skills |
| x | 7 | Unclear goals, Roles, or Responsibilities | | 7 | "Unsafe" Attitudes |
| xx | 8 | Lack of or Unclear Standards | | 8 | Illness/Fatigue (general health) |

[1] X = single occurrence, xx = multiple occurrences.

| WORK ENVIRONMENT (WE) | | | HUMAN NATURE (HN) | | |
|---|---|---|---|---|---|
| xx | 1 | Distractions/Interruptions | x | 1 | Stress |
| xx | 2 | Changes/Departure from Routine | | 2 | Habit patterns |
| | 3 | Confusing Displays/Controls | xx | 3 | Assumptions (inaccurate mental picture) |
| x | 4 | Work-arounds | | 4 | Complacency/overconfidence |
| xx | 5 | Hidden System/Equipment Response | x | 5 | Mindset (intentions) |
| xx | 6 | Unexpected Equipment Conditions | xx | 6 | Inaccurate Risk Perception |
| | 7 | Lack of Alternative Indication | | 7 | Mental Shortcuts (biases) |
| | 8 | Personality Conflicts | | 8 | Limited Short-term Memory |

**Task Demands**

There were several examples of a lack of clear standards, interpretation of requirements, and high work load that contributed to the severity of the incident. Lacking the establishment and reinforcement of clear standards and expectations, front line workers will establish their own standards of behavior based on their visions, beliefs, and values. The Operator did not have a clear understanding of expectations with regard to the use of the manual vehicle fire suppression system before the system.

**Work Environment**

There were numerous unexpected equipment conditions and equipment response encountered by the workers during this event (i.e., alarm not sounded for five seconds as expected, strobe lights not activated immediately, mine phone and pagers could not be heard throughout the mine, the manual fire suppression system did not fully actuate). These conditions affected the most effective and timely evacuation of the mine. Also, the manual vehicle fire suppression system could have eliminated the fire, or significantly slowed the progress of the fire.

**Individual Capabilities**

There were numerous issues related to individual capabilities in the area of proficiency, first time use, and a lack of knowledge for the intended task. There was no hands-on training in many areas necessary to provide worker proficiency. Several people had difficulty donning self-rescuers and SCSRS. The drill and exercises performed to date did not prepare individuals for this particular fire accident scenario. Inadequate guidance and training exists to support the FSM to make decisions without the requisite knowledge to fully understand the potential impact of the decision.

**Human Nature**

There were six different examples of Inaccurate Risk Perception error precursors on the part of personnel involved in the accident. Personnel that have an inaccurate risk perception typically base that on personal appraisal of hazards and uncertainty based on incomplete information or assumptions and/or an unrecognized or inaccurate understanding of a potential consequence or danger. The degree of risk-taking behavior is based on an individual's perception of the possibility of error and understanding of the consequences. There was an inaccurate risk perception on the part of FSM with regard to shifting ventilation modes.

**Questioning Attitude**

Individuals demonstrate a questioning attitude by challenging assumptions, investigating anomalies, and considering potential adverse consequences of planned actions. All employees must be watchful for conditions or activities that can have an undesirable effect on safety, and they do not proceed if faced with uncertainty. A reluctance to fear the worst is aggravated by human nature, since humans tend to accentuate the positive. A healthy questioning attitude must overcome the temptation to rationalize away something that is not right. A team approach where everyone is looking, questioning, and challenging every aspect of the work is required to increase the chances of identifying the job site hazards to ensure protection of the workers.

Based on interviews, there was little evidence that the workers displayed a questioning attitude. It was clear that if management has made a decision, workers do not challenge the decision.

## 9.4 Nuclear Culture and Mine Culture

### 9.4.1 Safety Culture

Production and prevention practices always compete in the minds of workers. Leaders have to constantly work hard to keep the facility, environment, and personnel safe. Well-informed leadership at all levels of the organization will ensure that the vision, beliefs, and values (prevention-centered attributes) do not conflict with the mission, goals, and processes (production-centered attributes). Consistency and alignment promote both production and prevention behaviors - together generating the desired long-term results.

In normal human behavior, production behaviors naturally take precedence over prevention behaviors unless there is a strong safety culture - nurtured by strong leadership. Sometimes managers err when they assume people will be or are safe. Safety and prevention behaviors do not just happen. They are value-driven, and people may not choose the conservative approach because of what is believed or perceived to be a stronger production focus.

It is critically important that the visions, values, and beliefs established by the leadership to support a strong safety culture are clearly communicated, and constantly reinforced. In many cases, management believes that their visions and values have been established and communicated through the development of a policy or procedure, or the posting of signs. That is an initial step and meets minimum compliance requirements, but it takes more than that. Leaders must constantly reinforce these expectations through observation and coaching at all levels of the organization.

Within DOE, most serious events do not occur when performing complex or high hazard operations. They rarely occur when starting up new facilities or performing operations for the first time. That is because everyone is paying close attention, there are lots of people involved, things move slowly, and everyone is very “mindful.” Natural tendency is to primarily focus on what are considered “high hazard” or “high risk” operations. The challenge for leadership is to establish and reinforce the safety culture expectations continuously so that workers are mindful and careful during all operations.

There are several examples concerning the accident where personnel “did not do” what was written down in a training briefing or what management expected them to do. There are several reasons for this, but foremost is a lack of strong safety expectations. The Board observed that there were examples of decisions regarding changes to equipment, maintenance of equipment, procedural compliance and CONOPS that were not conservative with respect to nuclear safety. A nuclear safety review of analyzed accidents with respect to the vehicle fire is provided to understand the expectations of maintaining underground vehicles not associated with waste handling.

The Documented Safety Analysis for the WIPP provides an analysis for a vehicle fire at the waste front. The “Single Liquid-Fueled Vehicle Collision and Fire at Waste Face Pool Fire”

bounds this type of accident. The analysis is developed based on a pool fire encompassing a contact-handled waste disposal array.

The following controls for reducing public risk from the hazardous conditions associated with Event CH-U/G-1-003a (single liquid-fueled vehicle collision and fire at waste face (pool fire)) have been identified as measures requiring inclusion in the TSR:

- U/G Liquid-fueled Waste-Handling Vehicles. The U/G liquid-fueled waste-handling vehicles are designed to prevent and/or mitigate fires.
- U/G Liquid-fueled Waste-Handling Vehicles Fire Suppression System. The U/G liquid-fueled waste-handling vehicles are equipped with a fire suppression system that suppresses fires associated with fuel line leaks and engine compartment fires.
- Vehicle/Equipment Control Program. Non-waste-handling vehicles are maintained greater than or equal to 25 feet from the waste when not attended.
- Liquid-fuel Vehicle/Equipment Inspection Program. Liquid-fueled vehicles/equipment approaching the waste face have pre-operational checks prior to use through the Underground Liquid-Fueled Vehicle/Equipment Inspection Program.

Limiting Condition for Operation 3.3.7, "Liquid-Fueled Vehicle/Equipment Control at a Waste Face," provides controls for limiting vehicles in the disposal room during activities. These controls include only waste-handling equipment selected for waste-handling activities during emplacement, limiting vehicles at the waste front during retrievals, and requirements to attend vehicles at the waste front or emplacing wastes. These controls are intended to ensure operation maintains the assumptions used in the safety analysis.

**Analysis**

The controls identified in the limiting condition for operations are intended to reduce the likelihood of fuel pool fires or accidents caused by facility equipment or improper equipment operation. Retrieval operations allow one non-waste-handling vehicle at the waste front in addition to one waste-handling vehicle. While there is a clear distinction in the analysis between waste-handling equipment and non-waste-handling (mining) equipment, the underlying assumption is that the non-waste-handling equipment is maintained in accordance with the checklists developed from the manufacturers.

However, the maintenance records and the removal of the automatic suppression from underground non-waste-handling vehicle/equipment do not reflect the degree of rigor necessary to assure that the nuclear safety basis and assumptions will be maintained. The condition of the vehicle in the fire challenges the integrity of the assumptions in the safety basis. The mine operations and nuclear operations underground are interrelated and need to be fully evaluated and better integrated.

**CON 21**: NWP and CBFO did not analyze and disposition differences between waste-handling and non-waste-handling vehicles for similar hazards and impacts, e.g., allowing a truck in this condition to be at the waste face.

**JON 34**: NWP and CBFO need to identify and control the risk imposed by non-waste-handling equipment, e.g., combustible buildup, manual vs. automatic fire suppression system, fire-resistant hydraulic oil, etc., or treat waste-handling equipment and non-waste-handling equipment the same.

**CON 22**: NWP and CBFO management allowed a culture to exist where there are differences in the way waste-handling equipment and non-waste-handling equipment are maintained and operated.

**JON 35**: NWP and CBFO management need to examine and correct the culture that exists regarding the maintenance and operation of non-waste handling equipment.

# 10.0 Analysis

## 10.1 Barrier Analysis

After a basic chronology of events was developed, the Board performed a barrier analysis of the accident. To start the barrier analysis, the Board chose a target (the person or item to be protected) and the hazard (what the person or item is to be protected from). The Board chose underground workers and facilities as the target and exposure to the fire and resultant smoke as the hazard. The Board also chose to include personnel evacuation and emergency response within the scope of the barrier analysis.

Thirty-eight barriers were identified and analyzed by the Board.

The barrier analysis is presented in Appendix B.

## 10.2 Change Analysis

To further support the development of causal factors, the Board performed a change analysis of the accident, examining the planned and unplanned changes that caused the undesired results or outcomes related to the event.

Thirty-nine changes were identified and analyzed by the Board.

The change analysis is presented in Appendix C.

## 10.3 Event and Causal Factors Analysis

After performing the barrier and change analyses, the Board assigned the results of the various analyses to the conditions that were related to or caused the events in the chronology. Correlating these conditions with events resulted in the events and causal factors chart provided in Appendix E. When the correlation was complete, the Board examined the chart to determine which events were significant (i.e., which events played a role in causing the accident). The Board then assessed the significant events (and the conditions of each) to determine the causal factors of the accident.

The causal factors that resulted are described below.

**Direct, Root, and Contributing Causes**

**Direct Cause** (DC) – the immediate events or conditions that caused the accident.

The Board identified the direct cause of this accident to be contact between flammable fluids (either hydraulic fluid or diesel fuel), and hot surfaces (most likely the catalytic converter) on the salt haul truck, which resulted in a fire that consumed the engine compartment and two front tires.

**Root Cause** (RC) – causal factors that, if corrected, would prevent recurrence of the same or similar accidents.

The Board identified the root cause of this accident to be NWP failure to adequately recognize and mitigate the hazard regarding a fire in the underground. This includes recognition and removal of the buildup of combustibles through inspections, and periodic preventative maintenance, e.g., cleaning and the decision to deactivate the automatic onboard fire suppression system.

**Contributing Causes** (CC) – events or conditions that collectively with other causes increased the likelihood or severity of an accident but that individually did not cause the accident. For the purposes of this investigation, contributing causes include those related to the cause of the fire as well as those related to the subsequent response.

The Board identified ten contributing causes to this accident or resultant response:

1. The preventative and corrective maintenance program did not prevent or correct the buildup of combustible fluids on the salt truck. There is a distinct difference between the way waste-handling and non-waste-handling vehicles are maintained.
2. The fire protection program was less than adequate in regard to flowing down upper-tier requirements relative to vehicle fire suppression system actuation from the Baseline Needs Assessment into implementing procedures. There was also an accumulation of combustible materials in the underground in quantities that exceeded the limits specified in the Fire Hazard Analysis (FHA) and implementing procedures. Additionally, the FHA does not provide a comprehensive analysis that addresses all credible underground fire scenarios including a fire located near the Air Intake Shaft.
3. The training and qualification of the operator was inadequate to ensure proper response to a vehicle fire. He did not initially notify the CMR that there was a fire or describe the fire's location.
4. The CMR response to the fire, including evaluation and protective actions, was less than adequate.
5. Elements of the emergency/preparedness and response program were ineffective.
6. A nuclear versus mine culture exists where there are significant differences in the maintenance of waste-handling versus non-waste-handling equipment.
7. The NWP Contractor Assurance System (CAS) was ineffective in identifying the conditions and maintenance program inadequacies associated with the root cause of this event.
8. DOE Carlsbad Field Office (CBFO) was ineffective in implementing line management oversight programs and processes that would have identified NWP CAS weaknesses and the conditions associated with the root cause of this event.
9. Repeat deficiencies were identified in DOE and external agencies assessments, e.g., Defense Nuclear Facility Safety Board (DNFSB) emergency management, fire protection, maintenance, CBFO oversight, and work planning and control, but were allowed to remain unresolved for extended periods of time without ensuring effective site response.

10. There are elements of the Conduct of Operations (CONOPS) program that demonstrate a lack of rigor and discipline commensurate with the operation of a Hazard Category 2 Facility.

The causal factors and related functions chart is presented in Appendix D.

The events and causal factors chart is presented in Appendix E.

# 11.0 Conclusions and Judgments of Need

Conclusions (CONs) are significant deductions derived from the investigation's analytical results. They are derived from and must be supported by the facts plus the results of testing and the various analyses conducted.

Judgments of Need (JONs) are the managerial controls and safety measures determined by the Board to be necessary to prevent or minimize the probability or severity of a recurrence. These JONs are linked directly to the causal factors which are derived from the facts and analysis. They form the basis for corrective action plans which must be developed by line management. The Board's conclusions and JONs are listed below in Table 5.

**Table 5: Conclusions and Judgments of Need**

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| **CON 1**: The FSM and Central Monitoring Room Operator (CMRO) did not fully follow the procedures for response to a fire in the U/G. This can be attributed to the complexity of the alarm and communication system, lack of effective drills and training, and additional burdens placed on the FSM due to the lack of a structured Incident Command System (ICS). | **JON 1:** NWP needs to evaluate and correct deficiencies regarding the controls for communicating emergencies to the underground, including the configuration and adequacy of equipment (alarms, strobes, and public address).<br>**JON 2**: NWP needs to evaluate the procedures and capabilities of the FSM and CMRO in managing a broad range of emergency response events through a comprehensive drill and requalification program. |
| **CON 2**: NWP management allows expert-based, rather than a process/systems-based approach to decision making, e.g., shift to filtration during a fire, sheltering decisions, etc. | **JON 3**: NWP needs to evaluate and apply a process/systems based approach for decision making relative to credible emergencies in the U/G, including formalizing response actions, e.g., decision to change to filtration mode during an ongoing evacuation. |
| **CON 3**: The emergency management program was not structured such that personnel were driven to adequately size up, properly categorize, and classify emergency events.<br>The WIPP (NWP and CBFO) emergency management program is not fully compliant with DOE O 151.1C, *Comprehensive Emergency Management System*, e.g., | **JON 4**: NWP and CBFO need to evaluate their corrective action plans for findings and opportunities for improvement identified in previous external reviews, and take action to bring their emergency management program into compliance with requirements.<br>**JON 5**: NWP and CBFO need to correct their activation, notification, classification, and categorization protocols to be in full |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| activation of the EOC, classification and categorization, emergency action levels, implementation of the ICS, training, triennial exercise, etc. Weaknesses in classification, categorization, and emergency action levels (EALs) were previously identified by external reviews and uncorrected. | compliance with DOE O 151.1C and then provide training for all applicable personnel.<br><br>**JON 6**: NWP and CBFO need to improve the content of site-specific EALs to expand on the information provided in the standard EALs contained in DOE O 151.1C.<br><br>**JON 7**: NWP and CBFO need to develop and implement an Incident Command System (ICS) for the EOC/CMR that is compliant with DOE O 151.1C and is capable of assuming command and control for all anticipated emergencies. |
| **CON 4**: Actions to be taken by the Operator in the event of a U/G vehicle fire were not clear.<br><br>There were inconsistencies between procedures and training for fire response that led to an ineffective response to the salt haul truck fire. | **JON 8**: NWP needs to review procedures and ensure consistent actions are taken in response to a fire in the U/G.<br><br>**JON 9**: NWP, CBFO and DOE HQ need to clearly define expectations for responding to fires in the U/G, including incipient and beyond incipient stage fires. |
| **CON 5**: NWP and CBFO failed to ensure that training and drills effectively exercised all elements of emergency response to include practical demonstration of competence, e.g., donning of self-rescuers and SCSRs, U/G personnel response to a fire, use of portable fire extinguishers, EOC roles, classification and categorization, notifications and reporting, and allowance of unescorted access for over 500 personnel, etc. | **JON 10**: NWP and CBFO need to develop and implement a training program that includes hands-on training in the use of personal safety equipment, e.g., self-rescuers, SCSRs, portable fire extinguishers, etc.<br><br>**JON 11**: NWP and CBFO need to improve and implement an integrated drill and exercise program that includes all elements of the ICS, including the MRT, First Line Initial Response Team (FLIRT) and mutual aid; unannounced drills and exercises; donning of self-rescuers/SCSRs; and full evacuation of the U/G.<br><br>**JON 12**: NWP needs to evaluate and improve their criteria for granting unescorted access to the U/G such that personnel with unescorted access to the underground are proficient in responding to abnormal events. |
| **CON 6**: NWP preventive and corrective maintenance program did not prevent or correct the buildup of combustible fluids on | **JON 13**: NWP management needs to reevaluate and modify the approach to conducting preventative and corrective maintenance on all U/G vehicles such that |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| the salt haul truck. | combustible fluids are effectively managed to prevent the recurrence of fires. |
| **CON 7**: NWP and CBFO management is not adequately considering overall facility impact with regard to operations, emergency response, and maintenance, which affects the safety posture of the facility, e.g., salt haul truck combustible build-up, conversion of the automatic fire suppression system to manual, removal of the automatic fire detection capability, not using fire resistant hydraulic fluid, discontinued use of the vehicle wash station, chaining of ventilation doors and an out-of-service regulator and fans, inoperable mine phones, and other non-waste-handling related equipment. | **JON 14**: NWP and CBFO need to develop and implement a rigorous process that effectively evaluates:<br>• changes to facilities, equipment, and operations for their impact on safety, e.g., plant operations review process;<br>• impairment and corresponding compensatory measures on safety-related equipment; and<br>• the impact of different approaches in maintaining waste-handling and non-waste-handling equipment.<br>**JON 15**: NWP needs to determine the extent of this condition and develop a comprehensive corrective action plan to address identified deficiencies. |
| **CON 8**: NWP and CBFO management have not effectively managed the quantity and duration of out-of-service equipment. | **JON 16**: NWP needs to develop and implement a process that ensures comprehensive and timely impact evaluation and correction of impaired or out-of-service equipment.<br>**JON 17**: CBFO needs to ensure that its contractor oversight structure includes elements for comprehensive and timely evaluation and correction of impaired or out-of-service equipment. |
| **CON 9**: NWP management has allowed less than acceptable rigor in the performance of equipment inspections, resulting in the operation of U/G equipment in unacceptable condition. | **JON 18**: NWP needs to develop and reinforce clear expectations regarding the performance of rigorous equipment inspections in accordance with manufacturer recommendations, established technical requirements; corrective action; and trending of deficiencies. |
| **CON 10:** NWP did not ensure the BNA addressed requirements of DOE O 420.1C and MSHA with the results completely incorporated into implementing procedures. | **JON 19**: NWP needs to ensure that all requirements of DOE O 420.1C and MSHA are addressed in the BNA with the results completely incorporated into implementing procedures and the source requirements |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| | referenced, and that training consistent with those procedures is performed. |
| **CON 11**: NWP and CBFO management did not make conservative or risk-informed decisions with respect to developing and implementing the fire protection program.<br><br>There is inadequate fire engineering analysis due to a lack of integration with ventilation design and operations, and U/G operations, for recognizing, controlling, and mitigating U/G fires. | **JON 20**: NWP and CBFO need to perform an integrated analysis of credible U/G fire scenarios and develop corresponding response actions that comply with DOE and MSHA requirements.<br><br>The analysis needs to include formal disposition regarding the installation of an automatic fire suppression system in the mine. |
| **CON 12**: NWP and CBFO have failed to take appropriate action to correct combustible loading issues that were identified in previous internal and external reviews. | **JON 21**: NWP and CBFO need to review the combustible control program and complete corrective actions that demonstrate compliance with program requirements. These issues remain unresolved from prior internal and external reviews. |
| **CON 13**: NWP and CBFO have allowed housekeeping to degrade and other conditions to persist that potentially impede egress. | **JON 22**: NWP and CBFO need to evaluate and address deficiencies in housekeeping to ensure unobstructed egress and clear visibility of emergency egress strobes, reflectors, SCSR lights, etc. |
| **CON 14**: NWP has not fully developed an integrated contractor assurance system that provides assurance that work is performed compliantly, risks are identified, and control systems are effective and efficient. | **JON 23**: NWP needs to develop and implement a fully integrated contractor assurance system that provides DOE and NWP confidence that work is performed compliantly, risks are identified, and control systems are effective and efficient. |
| **CON 15**: CBFO failed to adequately establish and implement line management oversight programs and processes to meet the requirements of DOE O 226.1B and hold personnel accountable for implementing those programs and processes. | **JON 24**: CBFO needs to establish and implement an effective line management oversight program and processes that meet the requirements of DOE O 226.1B and hold personnel accountable for implementing those programs and processes. |
| **CON 16**: CBFO management does not have adequate communication processes to ensure awareness of issues that warrant attention from all levels of the DOE staff. | **JON 25**: CBFO needs to accelerate the implementation of a mechanism for all levels of CBFO staff to document, communicate, track, and close issues both internally and with NWP. |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| | **JON 26**: The CBFO Site Manager needs to institutionalize and communicate expectations for the identification, documentation, reporting, and correction of issues. |
| **CON 17**: DOE HQ failed to ensure that CBFO was held accountable for correcting repeatedly identified issues involving fire protection, maintenance, emergency management, work planning and control, and oversight. | **JON 27**: DOE HQ needs to ensure that repeatedly identified issues related to safety management programs (SMPs) are confirmed closed and validated by the local DOE office.<br>This process should be considered for application across the DOE complex and include tracking, closure, actions to measure the effectiveness of closure (line management accountability), and trending to identify precursors and lessons learned.<br>**JON 28**: DOE HQ should enhance its required oversight to ensure site implementation of the emergency management policy and requirements are consistent and effective. Emphasis should be placed on ensuring ICSs are functioning properly and integrated exercises are conducted where personnel are evacuated. |
| **CON 18**: DOE HQ failed to ensure CBFO was provided with qualified technical resources to oversee operation of a Hazard Category 2 Facility in a mine. | **JON 29**: DOE HQ needs to develop and implement a process for ensuring that technical expertise is available to provide support in the unique area of ground control, underground construction, and mine safety and equipment.<br>**JON 30**: DOE HQ needs to assist CBFO with leveraging expertise from MSHA, in accordance with the DOE/MSHA MOU, in areas of ground control, underground construction, and mine safety where DOE does not have the expertise.<br>**JON 31**: DOE HQ needs to re-evaluate resources (i.e. funding, staffing, infrastructure, etc.) applied to the WIPP project to ensure safe operations of a Hazard Category 2 Facility. |
| **CON 19**: The Office of Environmental Management Consolidated Business Center (EMCBC) and CBFO failed to ensure | **JON 32**: EMCBC and CBFO need to develop and implement clear expectations and a schedule for EMCBC to provide support in the |

| Conclusion (CON) | Judgments of Need (JON) |
|---|---|
| support services as described in the Service Level Agreement were provided. | areas of regulatory compliance, safety management systems, preparation of program procedures and plans, quality assurance, lessons learned, contractor assurance, technical support, DOE oversight assistance, etc. |
| **CON 20**: There are elements of the CONOPS program that demonstrate a lack of rigor and discipline commensurate with operation of a Hazard Category 2 Facility. | **JON 33**: NWP and CBFO need to evaluate and correct weaknesses in the CONOPS program and its implementation, particularly with regard to flow-down of requirements from upper-tier documents, procedure content and compliance, and expert-based decision making. |
| **CON 21**: NWP and CBFO did not analyze and disposition differences between waste-handling and non-waste-handling vehicles for similar hazards and impacts, e.g., allowing a truck in this condition to be at the waste face. | **JON 34**: NWP and CBFO need to identify and control the risk imposed by non-waste-handling equipment, e.g., combustible buildup, manual vs. automatic fire suppression system, fire-resistant hydraulic oil, etc., or treat waste-handling equipment and non-waste-handling equipment the same. |
| **CON 22**: NWP and CBFO management allowed a culture to exist where there are differences in the way waste-handling equipment and non-waste-handling equipment are maintained and operated. | **JON 35**: NWP and CBFO management need to examine and correct the culture that exists regarding the maintenance and operation of non-waste-handling equipment. |
| **Positive Statement**: All supervisors and employees in the U/G actively used the mine phone to alert other workers of the fire and the need to evacuate before the evacuation alarm was sounded.<br><br>**Positive Statement**: Workers assisted other workers during the evacuation, including helping them to don self-rescuers and SCSRs.<br><br>**Positive Statement**: Personnel in the U/G exhibited detailed knowledge of the underground and ventilation splits.<br><br>**Positive Statement**: NWP on-site medical response was effective in treating personnel. | |

# 12.0 Board Signatures

Theodore A. Wyka
DOE Accident Investigation Board Chairman
U.S. Department of Energy, Office of Environmental Management

T.J. Jackson
DOE Accident Investigator and Deputy Chair
U.S. Department of Energy, Office of Environmental Management
Consolidated Business Center

Jack Zimmerman
DOE Accident Investigator and Board Member
U.S. Department of Energy, Office of Environmental Management
Portsmouth/Paducah Project Office

Roger Claycomb
DOE Accident Investigator and Board Member
U.S. Department of Energy, Office of Environmental Management
Idaho Operations Office

# 13.0 Board Members, Advisors and Consultants

**Board Members**

| | |
|---|---|
| Theodore A. Wyka | Board Chair, EM-40 Chief Nuclear Safety Advisor |
| T.J. Jackson | Board Deputy Chair, EMCBC, Trained Accident Investigator |
| Roger Claycomb | Board, ID, Trained Accident Investigator |
| Jack Zimmerman | Board, LEX, Trained Accident Investigator |

**Advisor/Team Coordinator**

| | |
|---|---|
| Advisor/Consultant | Greg Campbell<br>EMCBC, Emergency Management |
| Advisor/Consultant | Frank Moussa<br>EM-44, Emergency Management |
| Advisor/Consultant | James Landmesser<br>EM-41, Fire Protection |
| Advisor/Consultant | Ed Westbrook<br>EM-42, Work Controls |
| Advisor/Consultant | Jason Armstrong<br>Oakridge EM, Work Controls |
| Advisor/Consultant | Richard Lagdon,<br>EM-1, DOE HQ, Chief of Nuclear Safety |
| Advisor/Consultant | Micheal Ardaiz, MD, MPH, CPH,<br>DOE HQ, Chief Medical Officer |
| Advisor/Consultant | George Hellstrom,<br>CBFO, Legal Counsel |
| Advisor/Consultant | Randy Elmore<br>CBFO, Systems Engineering |
| Advisor/Consultant | Lina Pacheco<br>CBFO, Facility Representative |
| Advisor/Consultant | Don Galbraith<br>CBFO, Mine Ops Project Manager |
| Advisor/Consultant | Mark Williams<br>Supervisory Mine Safety and Health |

| | |
|---|---|
| Analyst/Advisor | Jack Gerber<br>MJW Technical Services |
| Analyst/Advisor | Robert Seal, MAS Consultants |
| Advisor/Consultant | Rick Callor, CSP<br>URS Professional Solutions, Boise |
| Advisor/Consultant | D. Allan Coutts, PE (SC), PhD, FSFPE<br>URS Professional Solutions, Aiken |
| Advisor/Consultant | Jim Stafford, CHP, PE, CSP<br>URS Professional Solutions |
| Observer | Todd Davis, DNFSB |
| Observer | Charles March, DNFSB |
| Administrative Coordinator/<br>Technical Writer | Susan M. Keffer, Project Enhancement Corporation<br>Trained Accident Investigator |

# Appendix A. Appointment of the Accident Investigation Board

**Department of Energy**
Washington, DC 20585

February 7, 2014

MEMORANDUM FOR GLENN S. PODONSKY
CHIEF HEALTH, SAFETY AND SECURITY OFFICER
OFFICE OF HEALTH, SAFETY AND SECURITY

FROM: MATTHEW MOURY
DEPUTY ASSISTANT SECRETARY
SAFETY, SECURITY, AND QUALITY PROGRAMS
ENVIRONMENTAL MANAGEMENT

SUBJECT: Determination of the Need to Conduct an Accident Investigation of the February 5, 2014 Fire Event at the Waste Isolation Pilot Plant

In accordance with the January 28, 2014, David Huizenga memorandum to Matthew Moury, *Delegation of Safety Authorities*, I have determined an Accident Investigation (AI) in accordance with Department of Energy Order 225.1B, *Accident Investigations,* is warranted for the February 5, 2014, fire at the Waste Isolation Pilot Plant in Carlsbad, New Mexico. Mr. Ted Wyka will be the Chairperson for the AI. He is assembling his AI team now and will be on site as soon as possible.

If you have any questions, please contact me or Mr. Ted Wyka, Chief Nuclear Safety Advisor, at (202) 287-5502.

cc: Jose Franco, WIPP
David Huizenga, EM-1
James Owendoff, EM-2 (Acting)
Jack Craig, EM-2.1 (Acting)
Frank Marcinowski, EM-30
James Hutton, EM-40
Ted Wyka, EM-40

# Appendix B. **Barrier Analysis**

Barrier analysis is based on the premise that hazards are associated with all tasks. A barrier is any means used to control, prevent, or impede a hazard from reaching a target, thereby reducing the severity of the resultant accident or adverse consequence. A hazard is the potential for an unwanted condition to result in an accident or other adverse consequence. A target is a person or object that a hazard may damage, injure, or fatally harm. Barrier analysis determines how a hazard overcomes the barriers, comes into contact with a target (e.g., from the barriers or controls not being in place, not being used properly, or failing), and leads to an accident or adverse consequence. The results of the barrier analysis are used to support the development of causal factors.

**Table B-1: Barrier Analysis**

| **Hazard: Fire and Smoke in the Underground** | | | **Target: Workers in the Mine** | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| Local Barriers for Preventing and/or Extinguishing the Salt Haul Truck Fire | | | | |
| **B1:** Onboard fire suppression system | Ineffective | Didn't fully discharge (no visible evidence) | Fire continued to burn | HPI:<br>WE-5,6<br>ISMS:<br>CF-3 |
| **B2:** Onboard portable fire extinguisher | Ineffective | Wasn't applied at source of the fire | Fire continued to burn | HPI:<br>WE-5,6<br>ISMS:<br>CF-3; GP-6 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B3:** Salt haul truck is designed to use non-flammable hydraulic fluid | Truck contains flammable hydraulic fluid | Did not analyze?<br>Is being used in the waste-handling vehicles. | May have contributed to fire | HPI:<br>N/A<br>ISMS:<br>CF-2; GP-1,5 |
| **B4:** 300 pound fire extinguisher | Ineffective | Unable to get to scene | Fire continued to burn | HPI:<br>IC-1, 3, 5<br>ISMS:<br>CF-1,3,4; GP-3,6 |
| **B5:** Rescue truck | Ineffective | Not used | Fire continued to burn | HPI:<br>WE-6<br>ISMS:<br>CF-2,3,4; GP-1,3,5,6 |
| **B6:** Maintenance/ housekeeping program for haul truck | Ineffective | Truck had accumulations of combustibles | Provided fuel source for fire | HPI:<br>TD-8,IC-2,HN-6<br>ISMS:<br>CF-2,3; GP-1,5,6 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B7:** Operator training for responding to fire on the truck | Partially effective | Did not call CMRO, did not activate onboard FPS first | Delayed application of fire suppression<br>Delayed response and evacuation | HPI:<br>IC-1,5 WE-5<br>ISMS:<br>CF-4; GP-5 |
| **B8:** Integrity of fluid systems | Ineffective (assumed) | Acceptance of leaks based on review of daily inspections and AIB walkdown of vehicles | There were fluid leaks | HPI:<br>HN-6,IC-2<br>ISMS:<br>CF-1,3,4; GP-1,5,6,7 |
| **B9:** Automatic detection and actuation of FPS | Ineffective | Removed due to inadvertent actuations | Was not applied until operator activated it | HPI:<br>TD-5,6; WE-2<br>ISMS:<br>CF-3,4; GP-1,5,6 |
| **B10:** Lessons learned from other fires, e.g, catalytic converter | Ineffective | Unaware (inadequacy in NWP Contractor Assurance System (CAS) – Lessons Learned (LL) program | LLs were not applied | HPI:<br>HN-6,3<br>ISMS:<br>CF-2,5; GP-1,6,7 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B11:** Self-assessment and oversight of haul truck condition | Ineffective | Identification and resolution of issues not performed adequately | Conditions were not identified | HPI:<br>N/A<br>ISMS:<br>CF-5; GP-1,5,6,7 |
| **B12:** Pre-operational checks | Ineffective | Performed but did not identify deficiencies | Combustibles were allowed to exist | HPI:<br>HN-3,6<br>ISMS:<br>CF-3,4,5; GP-1,5,6,7 |
| Local Barriers for Ensuring the Successful Evacuation of Personnel from the Underground After the Salt Haul Truck Fire | | | | |
| **B13:** Training and drills for underground fires | Partially effective | On POD, usually on Family day, no hands on practice, no integrated full scale exercise | Difficulties with donning both self-rescuers and SCSRs<br>Difficulties egressing | HPI:<br>WE-1,6; TD-1,IC-1,5, HN-1<br>ISMS:<br>CF-4,5; GP-3,5,6 |
| **B14:** PA system | Partially effective | Location, quantity, and volume of speakers inadequate | Inaudible in some areas or difficult to understand | HPI:<br>WE-6<br>ISMS:<br>CF-2,3; GP-3,5,6 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B15:** Alarms (yelp) | Partially effective | Did not alarm for 5 full seconds (CMRO action)<br>Do not ensure all mine phones are operable | Personnel trained to expect 5 second alarm<br>Not heard throughout the UG | HPI:<br>TD-2, WE-4,6<br>ISMS:<br>CF-4; GP-3,6 |
| **B16:** Strobes (evacuation lights) | Ineffective | Not turned on until called from UG<br>May not be operable throughout the UG (assumed)<br>Limited visibility (location) throughout the mine<br>Limited intensity (brightness)<br>Non uniform spacing | Difficulty in egress | HPI:<br>TD-2, WE-2,5, HN-1<br>ISMS:<br>CF-2,3,4; GP-2,6,7 |
| **B17:** Mine phones | Partially effective | May not be operable throughout the UG (run to failure) | Could not be heard throughout the UG, impeded understanding of fire | HPI:<br>N/A<br>ISMS:<br>CF-3; GP-6 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B18:** Mine reflectors | Partially effective | Some obscured by other equipment, mesh, salt dust, etc.<br>Not visible in heavy smoke<br>Non uniform spacing and heights | Impeded egress | HPI:<br>N/A<br>ISMS:<br>CF-3; GP-6 |
| **B19:** Ventilation (shift to filtration) | Partially effective | Counter to worker training on egress during evacuation<br>Contrary to industry practice<br>Contrary to step in UG Fire Response procedure<br>Not analyzed prior to event | Confusion in worker egress (smoke in areas expected to be safe) | HPI:<br>WE-2, HN-3,5,6, TD-6,7, IC-3,4<br>ISMS:<br>CF-2,3; GP-2,3,5,6,7 |
| **B20:** Ventilation control | Ineffective (inoperable for remote actuation) | Chained doors and regulator | Limited the options to control ventilation | HPI:<br>N/A<br>ISMS:<br>CF-3,5; GP-1,5,7 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B21:** Central Monitoring Room Operations | Partially effective | Did not follow procedures<br>Relied on FSM expertise and knowledge | Inadequate reporting and notifications<br>Confused workers UG | HPI:<br>WE-1,2, HN-3,6, TD-3,8, IC-2<br>ISMS:<br>CF-3,4; GP-1,2,3,5,6 |
| **B22:** Emergency Operations Center | Ineffective | Played no leadership role<br>No training for specific EOC position roles<br>Incident Command structure is not fully developed or implemented | Inadequate reporting and notifications<br>Failure to categorize<br>Failure to support the FSM by pushing resources | HPI:<br>N/A<br>ISMS:<br>CF-2,3,4; GP-1,2,3,5,6 |
| **B24**: Self-rescuers | Partially effective | No actual use (training) | Could not be donned by some personnel | HPI:<br>IC-5<br>ISMS:<br>CF-4; GP-3 |
| **B25:** SCSRs | Partially effective | No actual use (training) | Could not be donned by some personnel | HPI:<br>IC-5<br>ISMS:<br>CF-4; GP-3 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| Systemic Barriers for Ensuring the Safety of WIPP Personnel and the Environment | | | | |
| **B26:** Fire Protection Program | Ineffective – did not minimize the likelihood of the fire.<br>Program allowed:<br>• accumulation of combustibles in the vehicle near an ignition source.<br>• removal of automatic fire detection and suppression system (FPS) from truck. | Program addresses basic elements but BNA is less than adequate (previously identified by external reviews).<br>BNA states to contact CMRO, CMRO to dispatch UG Services to evaluate and fight fire, and then CMRO makes evacuation decision.<br>FHA did not evaluate a fire near a shaft underground.<br>Automatic FPS required for waste-handling vehicles but not for non-waste handling vehicles except if they are used near the waste face.<br>UG fire response procedure only | Uncontrolled fire in the underground.<br>Ineffective response to the fire.<br>Could cause significant delay in evacuation of the UG. | HPI:<br>N/A<br>ISMS:<br>CF-1,3,4; GP-1,2,5,6 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| | | addresses automatic FPS and use or a portable fire extinguisher | | |
| **B27:** Maintenance Program | Partially effective | Does not adequately consider management and control of combustibles.<br>Numerous red tagged fans, alarms, valve, pull station; some for greater than seven months.<br>Inoperable mine phones, possibly some strobes.<br>Inaudible public addresses system (in some locations). | Allowed fuel for fire.<br>No direct effect but reflects weakness in the program.<br>Inhibited egress.<br>Inhibited egress. | HPI:<br>N/A<br>ISMS:<br>CF-2,3; GP-1,2,5,7 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B28:** Emergency Management Program | Ineffective:<br>Lack of categorization.<br>Plays a support role to the CMR.<br>Failure to make some notifications.<br>Communications (yelp, strobes, status, direction). | Lack of rigor in training.<br>Procedures not followed.<br>Not specific enough (notifications, tactical support to FSM, response, emergency action levels).<br>Lack of defined roles for EOC staff.<br>No integrated annual exercise with external agencies.<br>Drills on schedule, typically performed on Wednesdays (Family Day) – no unannounced drills. | Ineffective command and control structure (CMR/EOC).<br>Delayed evacuation for some personnel. | HPI:<br>N/A<br>ISMS:<br>CF-2,3; GP-1,2,5,6,7 |
| **B29:** Underground Escape and Evacuation Plan | Effective | Did not fail. | No affect | HPI:<br>N/A<br>ISMS:<br>CF-2,3; GP-1,5,6,7 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B30:** Underground Fire Response Procedure | Ineffective:<br>Does not include BNA requirements to use the onboard manual FSP. | BNA requirements not included in the procedure. | Worker activated the FSP only after using the portable fire extinguisher (may have extinguished fire).<br>Workers were attempting to fight fire with the 300 lb extinguisher without sufficient hands on training. | HPI:<br>N/A<br>ISMS:<br>CF-1,2,3,4; GP-1,3,5,6 |
| **B31:** Training Program | Partially effective. | No hands on training (fire extinguishers, donning self-rescuers/SCSRs), drills and exercises didn't prepare UG personnel for this scenario | Some personnel did not follow procedures, drills and exercises were only partially effective, and some personnel encountered difficulties donning and wearing self-rescuers and SCSRs | HPI:<br>IC-1,2<br>ISMS:<br>CF-1, CF-4, GP-3 |
| **B32:** Documented Safety Analysis Program/Technical Safety Requirements | Effective | Did not fail | Did not – accident is bounded. | HPI:<br>N/A<br>ISMS:<br>GP-5 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B33:** Ventilation System Control | Ineffective: Distributed smoke throughout the UG. | Fire at this location had not been analyzed to take the appropriate ventilation control actions to minimize and/or eliminate smoke in the UG (similar to MSHA requirement for fire in an intake shaft). | Distributed smoke throughout the UG. | HPI:<br>N/A<br>ISMS:<br>CF-2,3; GP-1,5,6,7 |
| **B34:** Medical Response | Effective | Did not fail | Timely and efficient | HPI:<br>N/A<br>ISMS:<br>N/A |
| **B35:** Contractor Assurance System | Ineffective | Does not have | | HPI:<br>N/A<br>ISMS:<br>CF-5; GP-1,2,3,5,7 |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B36:** DOE CBFO | Partially effective | Inadequate resolution of externally identified issues<br>Emergency mgt. assessment triennial assessment has not been performed.<br>FR program assessment<br>FR/SME communication<br>Facility Representative program:<br>Staffing (only 2 of 4 FRs) due to medical issues.<br>In-development.<br>No structured surveillance program. | Did not identify issues with ineffective or failed barriers identified in this analysis. | HPI:<br>N/A<br>ISMS:<br>CF-5; GP-1,2,5,7 |
| | | Informal documentation and tracking of issues | | |

| Hazard: Fire and Smoke in the Underground | | | Target: Workers in the Mine | |
|---|---|---|---|---|
| **Barriers** | **How did barrier perform?** | **Why did barrier fail?** | **How did barrier affect accident?** | **Context: HPI/ISMS** |
| **B37:** External Oversight (DOE HQ, EMCBC, MSHA, DNFSB, etc.) | Partially effective | Acceptance or lack of enforcement of inadequate development and/or implementation of corrective actions to issues identified by these organizations. | Allowed for long-standing deficiencies in emergency management, fire protection, oversight, etc. to remain unresolved for extended periods | HPI:<br>N/A<br>ISMS:<br>CF-5; GP-1,2,5,6,7 |
| **B38:** Response to external Oversight (DOE HQ, EMCBC, MSHA, DNFSB, etc.) | Ineffective | There is ineffective site (CBFO and NWP) response (corrective action) to issues identified by these organizations. | Allowed for long-standing deficiencies in emergency management, fire protection, oversight, etc. to remain unresolved for extended periods | HPI:<br>N/A<br>ISMS:<br>CF-5; CP-1,2,5,6,7 |

# Appendix C. **Change Analysis**

Change is anything that disturbs the "balance" of a system from operating as planned. Change is often the source of deviations in system operations. Change can be planned, anticipated, and desired, or it can be unintentional and unwanted. Change analysis examines the planned or unplanned disturbances or deviations that caused the undesired results or outcomes related to the accident. This process analyzes the difference between what is normal (or "ideal") and what actually occurred. The results of the change analysis are used to support the development of causal factors.

**Table C-1: Change Analysis**

| Accident Situation | Prior, Ideal or Accident-Free Situation | Difference | Evaluation of Effect |
|---|---|---|---|
| Local Changes for Preventing and/or Extinguishing the Salt Haul Truck Fire | | | |
| C1: Mine atmosphere unsafe | Mine atmosphere safe | Significant smoke in the underground | Smoke inhalation, difficulty evacuating |
| C2: Fire and smoke in underground | No fire or smoke in underground | Significant smoke in the underground | Smoke inhalation, difficulty evacuating, soot on equipment and mine, soot on pre-filters |
| C3: Combustible fluid leaks on underground vehicles | No or minimal combustible fluid leaks on underground vehicles | Combustible fluids were available to combust.<br>Maintenance program ineffective or not followed. | The combustible fluid ignited when in contact with hot surfaces of the salt haul truck |
| C4: The on-board fire suppression system required activation by the salt haul truck driver. | The automatic fire suppression system activates at first indication of fire. | Delay in the time for activation of the on-board fire suppression system. | Fire may have been extinguished while in the incipient stage. |

| Accident Situation | Prior, Ideal or Accident-Free Situation | Difference | Evaluation of Effect |
|---|---|---|---|
| Local Barriers for Ensuring the Successful Evacuation of Personnel from the Underground After the Salt Haul Truck Fire | | | |
| C5: Issues donning self-rescuers and self-contained self-rescuers (SCSR) | No issues donning self-rescuers or SCSRs | Some personnel did not wear self-rescuers.<br>Training ineffective or inadequate. | Smoke inhalation |
| C6: Emergency alarm short, not heard everywhere | Emergency alarm for 5 seconds as per training and heard throughout the underground | Not all personnel were aware of the need to evacuate.<br>CMR did not leave yelp alarm for standard 5 seconds. | Delay in evacuation |
| C7: Emergency strobes not turned on or not visible throughout underground | Emergency strobes turned on at same time as "yelp" (or directly thereafter), remain on, and are visible throughout the underground | Not all personnel were aware of the need to evacuate | Delay in evacuation |
| C8: Personnel did not don self-rescuers at first sign of smoke | Personnel don self-rescuers at first sign of smoke | Not all personnel were wearing self-rescuers as required | Potential for smoke inhalation |
| C9: Announcements not audible and/or clear and not heard throughout the underground | Announcements were clear and concise and were heard throughout the underground | Not all personnel were aware of the need to evacuate and/or where the fire was located.<br>Public Address (PA) system ineffective. | Delay in evacuation and inability to plan best exit route |

| Accident Situation | Prior, Ideal or Accident-Free Situation | Difference | Evaluation of Effect |
|---|---|---|---|
| C10: Personnel were preparing to fight the fire wearing self-rescuers | Personnel did not wear self-rescuers to fight the fire | Personnel were preparing to fight the fire wearing only self-rescuers.<br>Training ineffective. | Potential for smoke inhalation |
| C11: Decision to change to filtration during the fire made based on personal experience | Decision on changes to filtration during a fire is based on analysis and full understanding of consequences. | Significant build-up of smoke in the mine and smoke in areas personnel expected to have "good air".<br>Experienced based decision making was inadequate. | Delay in evacuation, potential for personnel to become incapacitated during travel to the waste hoist |
| C12: Near-misses when driving/walking to waste hoist for evacuation | No near-misses when driving/walking to the waste hoist for evacuation | A number of near-misses (collisions) with people, carts, and/or equipment occurred.<br>Housekeeping less than adequate.<br>No designated travel paths.<br>No lights/strobes, or adequate reflectors on equipment. | Potential for personnel injuries and blockages to egress |
| C13: Hoist not at bottom when evacuation began. | Hoist "parked" at bottom when not in use. | Hoist wasn't available to immediately evacuate personnel. | Slight delay in evacuation. |
| C14: Manual fire suppression system was activated late in the fire | Automatic FSS that functions as designed and extinguishes fire in the incipient state | Significant reduction in the time the suppressant was applied | Fire didn't get extinguished in the incipient state |

| Accident Situation | Prior, Ideal or Accident-Free Situation | Difference | Evaluation of Effect |
|---|---|---|---|
| C15: Operator called maintenance and then his Supervisor about the fire | Operator calls the CMRO to report the fire | Delay in CMRO notification | Slight delay in reporting and evacuation |
| C16: Operator uses portable fire extinguisher first | Operator activates FSS first then use portable fire extinguisher | Significant reduction in the time the suppressant was applied | Fire didn't get extinguished in the incipient state |
| C17: Combustible fluids buildup on the salt haul truck | Combustible fluid managed in accordance with the owner's manual | Combustible fluids were available to ignite | Combustible fluids ignited |
| C18: Pre-operational checks on salt haul truck did not identify fluid buildup | Pre-operational check identifies fluids and has them addressed | Combustible fluids were available to ignite | Combustible fluids ignited |
| C19: Salt dump area and travel path is adjacent to and in the primary air intake split | Salt dump area and travel path is away from the air intake split | Smoke was distributed both north and south | Impeded egress |
| C20: UG Services responds to fire with only their self-rescuers | Trained fire response with proper PPE and firefighting equipment or clear policy to immediately evacuate | UG Services personnel not prepared to fight fire | UG Services personnel at risk |
| C21: Yelp was shorter in activation than required, delay in activating strobes, inaudible in some areas | Prompt activation of yelp alarm and strobes, audible and clear communication of instructions | Confusion in identifying type of emergency (or if a drill), expected response, and egress | Impeded egress |

| Accident Situation | Prior, Ideal or Accident-Free Situation | Difference | Evaluation of Effect |
|---|---|---|---|
| C22: Not all personnel donned self-rescuers at first indication of fire | All personnel don self-rescuers at first sign of fire | Some personnel were in smoke without wearing self-rescuers | Smoke inhalation and inability to evacuate |
| C23: Some personnel couldn't open and don their self-rescuers and SCSRs | Personnel have no difficulty opening and donning their self-rescuers and SCSRs | Some personnel were exposed to greater amount of smoke | Potential smoke inhalation |
| C24: CMR changed the ventilation to filtration during the incident | Should have followed their procedure and not switched to filtration (or come out of filtration if in that mode) | Potentially effected the locations of good air and concentrations of CO, put smoke in areas workers have been trained and expected to have good air | Spread smoke , confused workers, delayed egress |
| C25: Personnel were not prepared via drills and exercises for scenario where all underground has smoke | Drills and exercises prepare personnel for a scenario where all the UG is filled with smoke | Personnel were surprised and unprepared for situation and had to develop own egress plans | Delay in egress or failure to egress |
| C26: Alarms and announcements not heard throughout the UG, | Alarms and communication equipment operates as expected | Personnel not aware of need to evacuate or instructions | Delay in egress or failure to egress |
| C27: Strobes may not have been operable or visible throughout the mine | Strobes turned on with yelp alarm and are visible throughout the UG | Personnel could not see strobes to assist in egress | Delay in egress or failure to egress |

| Accident Situation | Prior, Ideal or Accident-Free Situation | Difference | Evaluation of Effect |
|---|---|---|---|
| C28: Did not categorize or make notifications | EOC activation, categorization, and notifications made in accordance with DOE O 151.1C | Event not properly categorized and required notifications were not made | Didn't trigger support from DOE and external agencies. |
| C29: FSM controlled all actions | Crisis manager with EOC support controls all actions allowing the FSM to focus on operational response | Decisions are made with limited input and support, potential for overload of FSM | Potential for inadequate response to the accident |
| Systemic Changes for Ensuring the Safety of WIPP Personnel and the Environment | | | |
| C30: BNA and FHA did not address this scenario, specifically the location of the fire (didn't consider fire in the supply drifts) | Fire Protection Program is effective. BNA and FHA analyze and pre-plan for credible scenarios. | No detailed analysis and response to this scenario | FSM had to develop response (location specific) at time of crisis |
| C31: Salt truck had combustible buildup, alarms could not be heard or understood throughout the mine, reflectors were not able to be seen during egress. | Maintenance Program effective. Equipment is properly maintained (trucks, alarms, strobes, PA system, reflectors, mine phones and pager). | Equipment was not effective in notifying personnel UG of the fire or expected evacuation | Delay in egress or potential for failure to egress |

| Accident Situation | Prior, Ideal or Accident-Free Situation | Difference | Evaluation of Effect |
|---|---|---|---|
| C32: Some red tagged safety related equipment for over 7 months. | Minimal backlog of impaired/inoperable safety related equipment with short period impaired/inoperable. | There is a relatively high number of impaired/out-of-service safety related equipment that has been in that state for an extended period of time. | No direct impact on this event. None of the red tagged equipment was relative to the fire or response. |
| C33: Lack of an integrated emergency management system. | Emergency Management Program effective. System is integrated with offsite agencies, site complies with requirements for categorization and notification. | Plans were not followed. | Not using all resources that are available |
| C34: Does not consider manual only initiation of onboard fire suppression systems.<br>Did not flow BNA requirements. | Underground Fire Response Procedure effective. Directs activation of manual fire suppression system upon discovery of vehicle fire. | Not instructed to initiate fire suppression system first. | May extinguish fire at incipient stage. |
| C35: A fire in the non-waste handling section of the mine adversely affected the ventilation system, including smoke and soot on the HEPA filtration system, waste handling building, and waste hoist. | Documented Safety Analysis Program/Technical Safety Requirements effective and includes evaluation of impacts of non-waste incidents, e.g., fire with a salt truck that impacts ventilation system. | Non-waste handling equipment is treated differently than waste handling equipment. | There was an impact on the HEPA filtration system from a fire involving a non-waste handling vehicle, waste handling building, and waste hoist. |

| Accident Situation | Prior, Ideal or Accident-Free Situation | Difference | Evaluation of Effect |
|---|---|---|---|
| C36: Some ventilation doors were chained open and a regulator was not operating properly.<br>In filtration mode, there is one door that must be manually positioned. | Ventilation System Control effective<br><br>All ventilation doors and regulators can be operated automatically or remotely. | Limits options for automatically or remotely controlling ventilation flow paths. | Inability to control the flow of smoke and cannot recover from filtration mode. |
| C37: Fire protection, emergency management, and CBFO oversight issues identified by DOE and external agencies have not been addressed and/or resolved in a timely manner | Complete and prompt response and resolution of issues identified by DOE and external agencies. | Could have had a fully compliant and effective fire protection, emergency management, and CBFO oversight program. | May have identified precursors to this incident. |
| C38: FR program is currently understaffed, no schedule for surveillances, issues are not documented and tracked through closure | DOE CBFO oversight (SME, FRs) is structured, fully staffed, and effective. | Could have identified conditions or inadequacies that caused this event. | Did not identify precursors to this incident. |
| C39: Contractor Assurance System did not identify conditions or precursors to this event. | Contractor Assurance System is effective – staffed, self-assessment and oversight is performed, issues are addressed, trending is performed | Could have identified conditions or inadequacies that caused this event | Did not identify precursors to this incident. |

# Appendix D. Causal Factors and Related Conditions

**Table D-1: Causal Factors and Related Conditions**

| Causal Factor | Related Conditions |
|---|---|
| **C1**: The preventative and corrective maintenance program did not prevent or correct the buildup of combustible fluids on the salt truck. | Buildup of combustibles on the salt haul truck.<br>Vehicle washing station was removed from service. Vehicle service manual requires washing every 100 hours of operation or every two weeks.<br>Difference in expectations for waste-handling vs non-waste-handling vehicles.<br>Unclear if compensatory measures for impaired safety related equipment have been identified or are in-place. (CONOPS)<br>Numerous mine phones were inoperable (run to battery failure). Twelve of the 40 tested were non-functional.<br>PA announcements were difficult to hear or understand.<br>Salt haul trucks are designed and built to use fire resistant hydraulic fluid, but it is not used in these vehicles.<br>Ability to change ventilation configuration remotely to control smoke is hampered by chained doors and a regulator in need of repair.<br>Expectations for clearing red tags on critical safety equipment, e.g., fans, fire suppression system, CAMs. No method to readily understand status of impaired safety related equipment. (CONOPS) |

| Causal Factor | Related Conditions |
|---|---|
| **C2**: Fire protection program less than adequate. | The BNA requirement for use of manual onboard FSS before use of portable fire extinguisher not included in U/G fire response procedure and therefore the onboard FSS was not activated first.<br><br>Decision to disable the automatic fire suppression system due to inadvertent actuation (engineering).<br><br>BNA has long-standing open issues, e.g., evaluation of the needs for U/G firefighting activities.<br><br>The U/G fire response procedure requires the CMRO to direct U/G Services to respond and evaluate fires after decision to evacuate the mine.<br><br>FHA did not consider the impact of a vehicle fire in this location.<br><br>MSHA requires evaluation and control of fires at a wooden shaft, this event simulates a fire at a wooden shaft and no evaluation has been performed or controls have been specified.<br><br>No direct relationship between the Fire Hazard Analysis (FHA) and this event.<br><br>Conditions in the U/G exceeded combustible loading limits during the event. |
| **C3**: CMR response (evaluation and protective actions) was less than adequate. | Did not sound emergency evacuation alarm for the full 5 seconds as required by procedure.<br><br>Did not immediately activate emergency strobe lights until notified by personnel U/G (~ 4 – 5 minute delay).<br><br>Unreasonable expectations and uncertain capabilities of the FSM to directly manage all aspects of an emergency abnormal event.<br><br>Alarm and communication system (control box) is not user friendly, e.g., strobes must be activated independent of the alarms and independent of the PA.<br><br>There is no longer a training week built into the CMR rotation schedule. |

| Causal Factor | Related Conditions |
|---|---|
| **C4**: Training and qualification of the CMR operator was inadequate to ensure proper response to a vehicle fire. | Did not sound emergency evacuation alarm for the full 5 seconds as required by procedure.<br><br>Did not immediately activate emergency strobe lights until notified by personnel U/G (~ 4 – 5 minute delay).<br><br>Unreasonable expectations and uncertain capabilities of the FSM to directly manage all aspects of an emergency abnormal event.<br><br>Alarm and communication system (control box) is not user friendly, e.g., strobes must be activated independent of the alarms and independent of the PA.<br><br>There is no longer a training week built into the CMR rotation schedule. |
| **C5**: Elements of the emergency management/preparedness and response program were ineffective. | Buildup of debris on reflectors, covered reflectors, blocked reflectors, irregular spacing of reflectors compounded the difficulty in egress due to the heavy smoke.<br><br>There were equipment and materials in the drifts that also made egress difficult and resulted in near-misses (collisions with people and equipment) in the heavy smoke.<br><br>Inconsistency between site EM program and DOE O 151.1C with regard to activation of the EOC.<br><br>Failure to classify and categorize.<br><br>Failure to make required notifications and reports.<br><br>No integrated emergency management program (notification, classification, and categorization).<br><br>No implementation of the ICS system between the scene of the accident, the EOC, and DOE HQ.<br><br>The EOC does not play a leadership role, the CMR maintains command of the event.<br><br>Incident command structure is not fully developed or implemented.<br><br>Some FSMs do not have the ICS series training.<br><br>No training for specific EOC position roles. |

| Causal Factor | Related Conditions |
|---|---|
| | No unannounced drills (on schedule, usually on Family Day)<br><br>No fully integrated exercises where personnel are fully evacuated and offsite agencies respond, e.g., MSHA, other than notifications.<br><br>A triennial emergency management self-assessment has not been conducted since 2008 and maybe not at all.<br><br>Effectiveness of training in donning and use of self-rescuers and SCSRs (many had trouble with one or both). Annual refresher is a video with no donning and therefore no evaluation of competency.<br><br>Rigor of training for salt truck drivers (used portable first instead of FSS).<br><br>Not all personnel receive hands on training on portable fire extinguisher use.<br><br>There are currently over 500 personnel granted unescorted access to the U/G. Many of these individuals have little familiarity with the U/G or evacuation expectations. |
| **C6**: Nuclear versus mine culture. | Different treatment of waste vs non-waste handling equipment, e.g., combustible buildup, manual vs. automatic FSS, fire resistant hydraulic oil, etc.<br><br>DSA/TSR LCO 3.3.7 allowed this truck in this condition to be at the waste face.<br><br>There is a difference in the level of oversight and attention on WH vs non-WH equipment. |
| **C7**: There are elements of the Conduct of Operations program that demonstrate a lack of rigor and discipline commensurate with the operation of a Hazard Category 2 Facility. | Maintenance Procedure PM074080, EMCO Haul Truck 74-U-006A/B, does not refer to the CHAMPS Preventative Maintenance process, nor include performance requirements from manufacturer's instructions. While "Various O&M Manuals" are listed as a reference in the procedure, there are no steps in the procedure that direct the user to refer to the manufacturer's instructions and validate performance criteria.<br><br>Operator's Checklists are not completely filled |

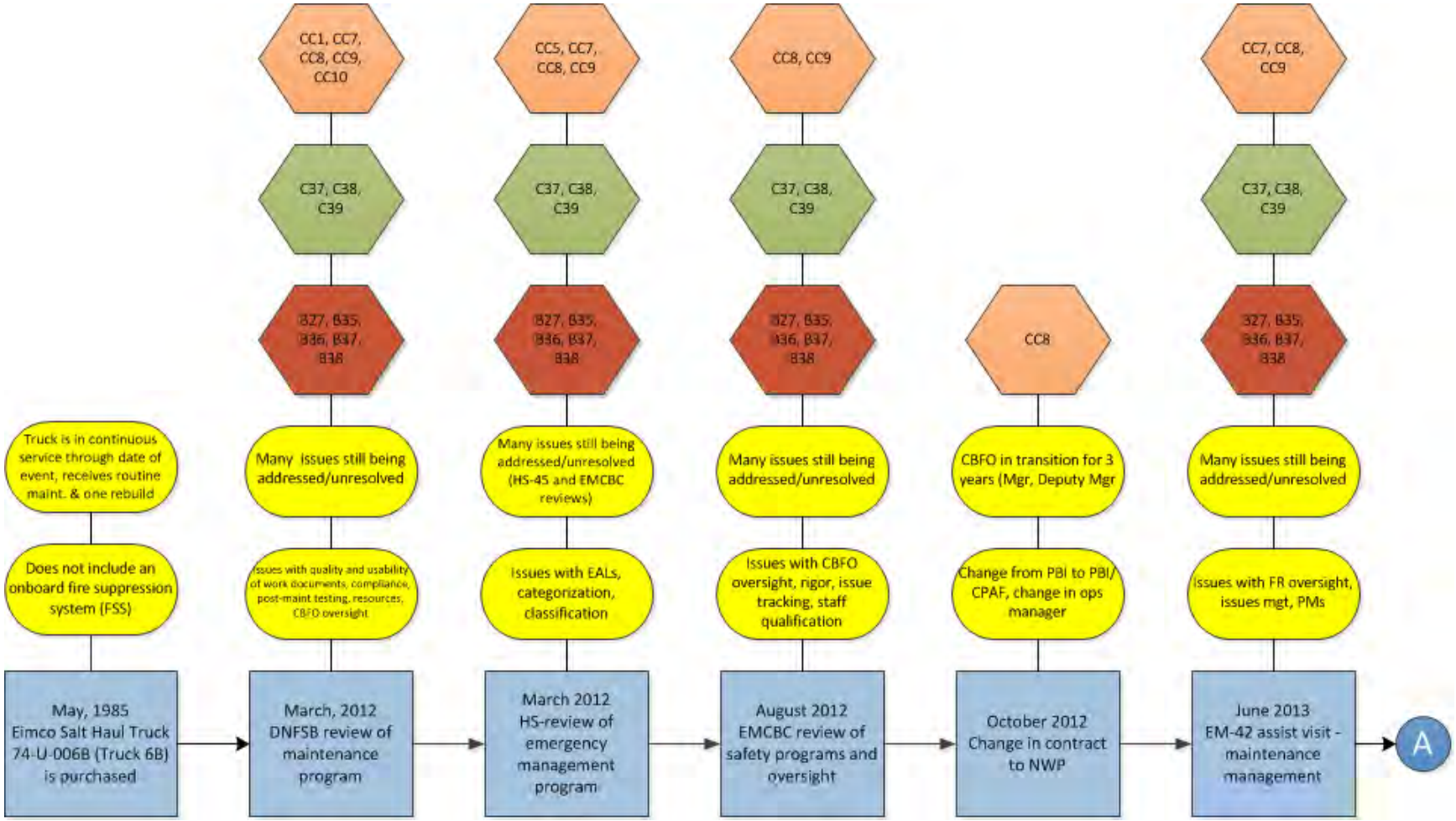
| Causal Factor | Related Conditions |
|---|---|
| | out. On several occasions, the initial and/or final meter reading was not filled out, and the machine (Haul Truck) is not marked as safe to use.<br><br>The emergency response procedures did not clearly identify points when U/G ventilation should be secured and/or changed, egress methods for conditions when multiple people are in the U/G, or when to activate the EOC.<br><br>The BNA requirement for use of the manual onboard FSS before use of a portable fire extinguisher was not included in the U/G fire response procedure.<br><br>The U/G fire response procedure required the CMRO to direct U/G Services to respond and evaluate fires after a decision was already made to evacuate the mine.<br><br>As identified in training and written in procedures, the haul truck operator did not notify the CMR of the fire after the portable fire extinguisher and manual FSS failed. The operator contacted the maintenance department.<br><br>Although required by the evacuation procedure, the CMR did not sound the evacuation alarm for a full 5 seconds and illuminate the emergency strobe lights.<br><br>Many U/G personnel were unable to don SRs and/or SCSRs.<br><br>Critical safety equipment had red tags in which NWP employees via interviews did not fully understand the status of the impaired safety related equipment. Safety equipment included fans, FSS, and CAM.<br><br>The U/G vehicle operator did not receive hands-on training on the use of portable fire extinguishers. |
| **C8**: NWP Contractor Assurance System (CAS) was ineffective. | Did not identify precursors through self-assessment or independent oversight.<br><br>Did not identify and disseminate pertinent lessons learned.<br><br>Ineffective corrective action to externally |

| Causal Factor | Related Conditions |
|---|---|
| | identified issues.<br>Management walkdowns of the U/G (if/when performed) did not identify conditions causal to the fire, housekeeping, combustible loading, mine phone inoperability, etc.<br>External organizations identify issues not pre-identified through NWP self-assessment and/or oversight |
| **C9**: CBFO oversight was ineffective. | Did not identify precursors through oversight, i.e., Facility Representative program or oversight.<br>Inadequate management attention, tracking and trending, and execution of the WIPP corrective action program.<br>Lost opportunities to utilize MSHA inspections and assist visits required by public law and the MOU.<br>Facility Representative program is ineffective:<br>• Procedures incomplete<br>• Staffing does not meet staffing analysis<br>• No structured surveillance program.<br>Inadequate communication of issues to DOE and contractor management.<br>FR/SME communications/barriers. |
| **C10**: Repeat deficiencies were identified in DOE and external agency assessments, e.g., DNFSB emergency management, fire protection, maintenance, CBFO oversight, and work planning and control, but allowed to remain unresolved for extended periods of time without ensuring effective site response. | There are numerous issues from DOE HQ, EMCBC, and the DNFSB which remain unresolved and have been so for extended periods of time. |

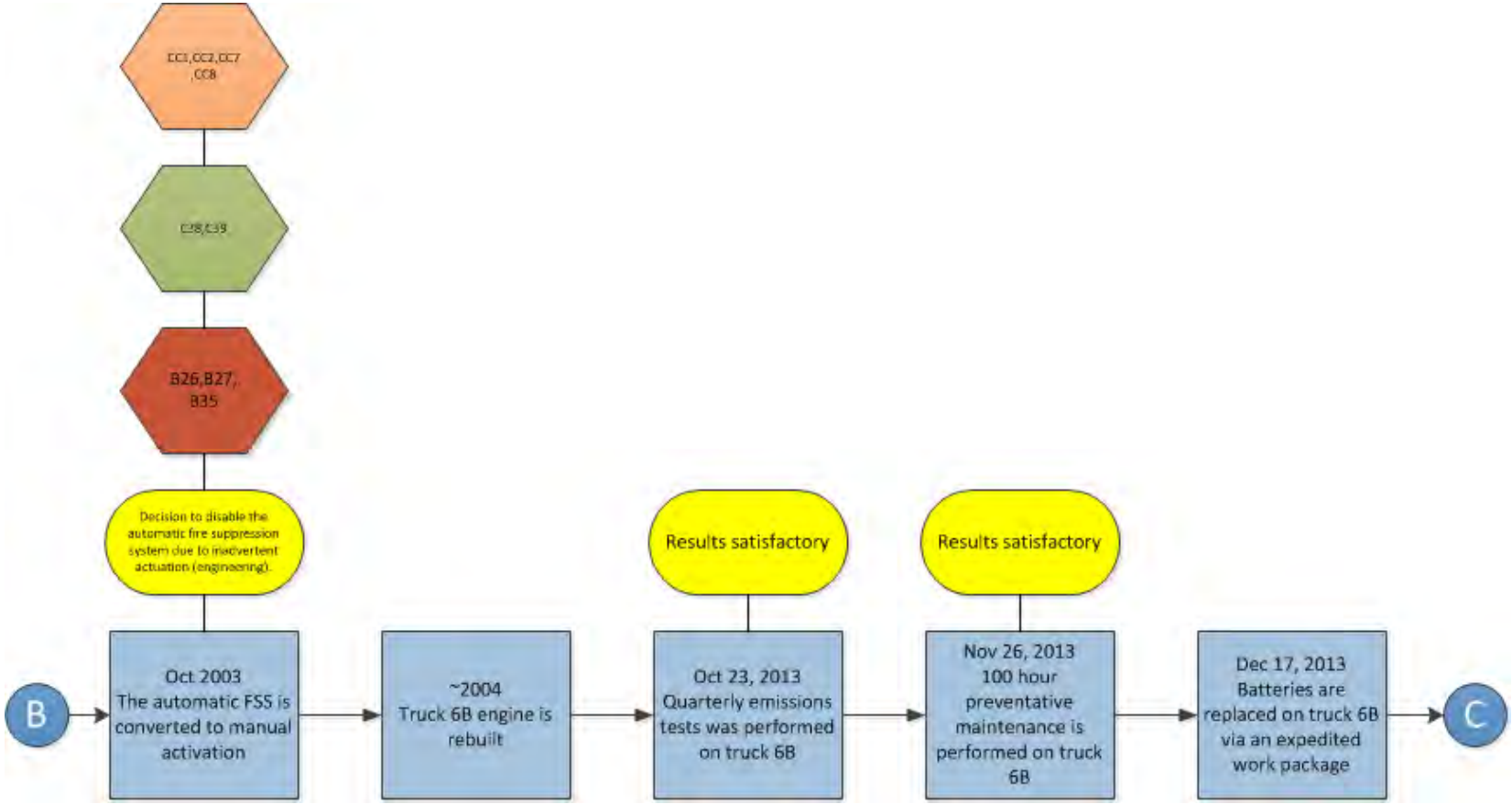
# Appendix E. **Event and Causal Factor Analysis**

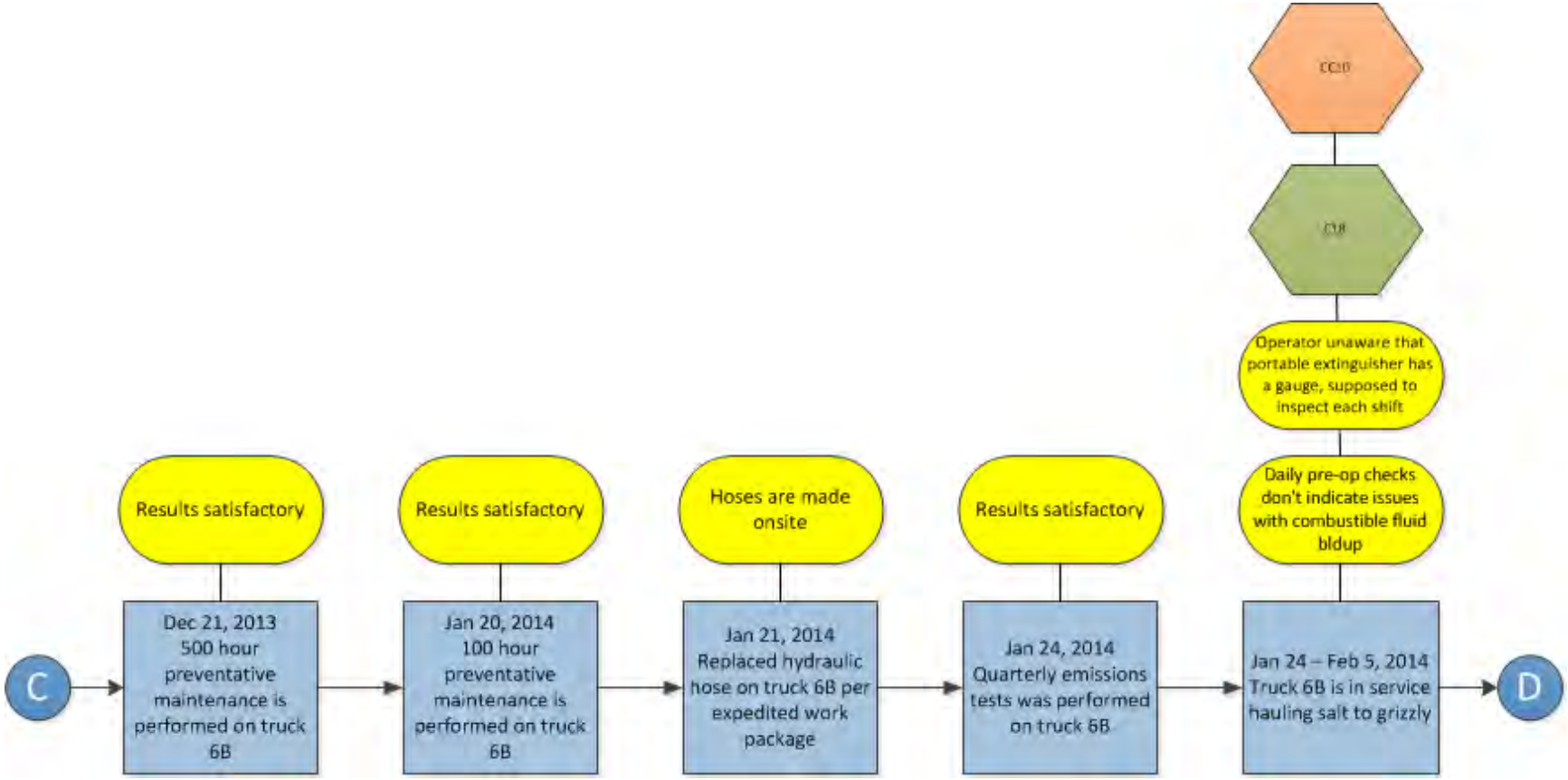
An events and causal factors analysis was performed in accordance with the DOE Workbook, Conducting Accident Investigations. The events and causal factors analysis requires deductive reasoning to determine those events and/or conditions that contributed to the accident. Causal factors are the events or conditions that produced or contributed to the accident, and they consist of direct, contributing, and root causes. The direct cause is the immediate event(s) or condition(s) that caused the accident. The contributing causes are the events or conditions that, collectively with the other causes, increased the likelihood of the accident, but which did not solely cause the accident. Root causes are the events or conditions that, if corrected, would prevent recurrence of this and similar accidents. The causal factors are identified in Figure D-1: Events and Causal Factors Analysis.

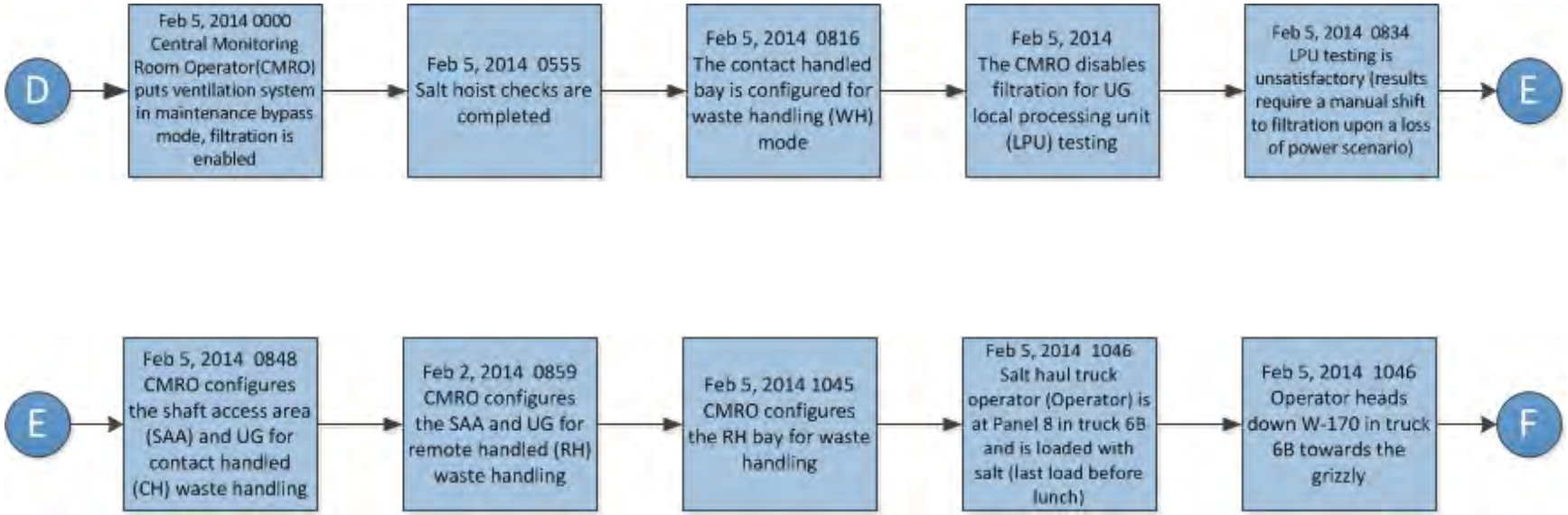
**Table D-1: Event and Causal Factors Analysis**

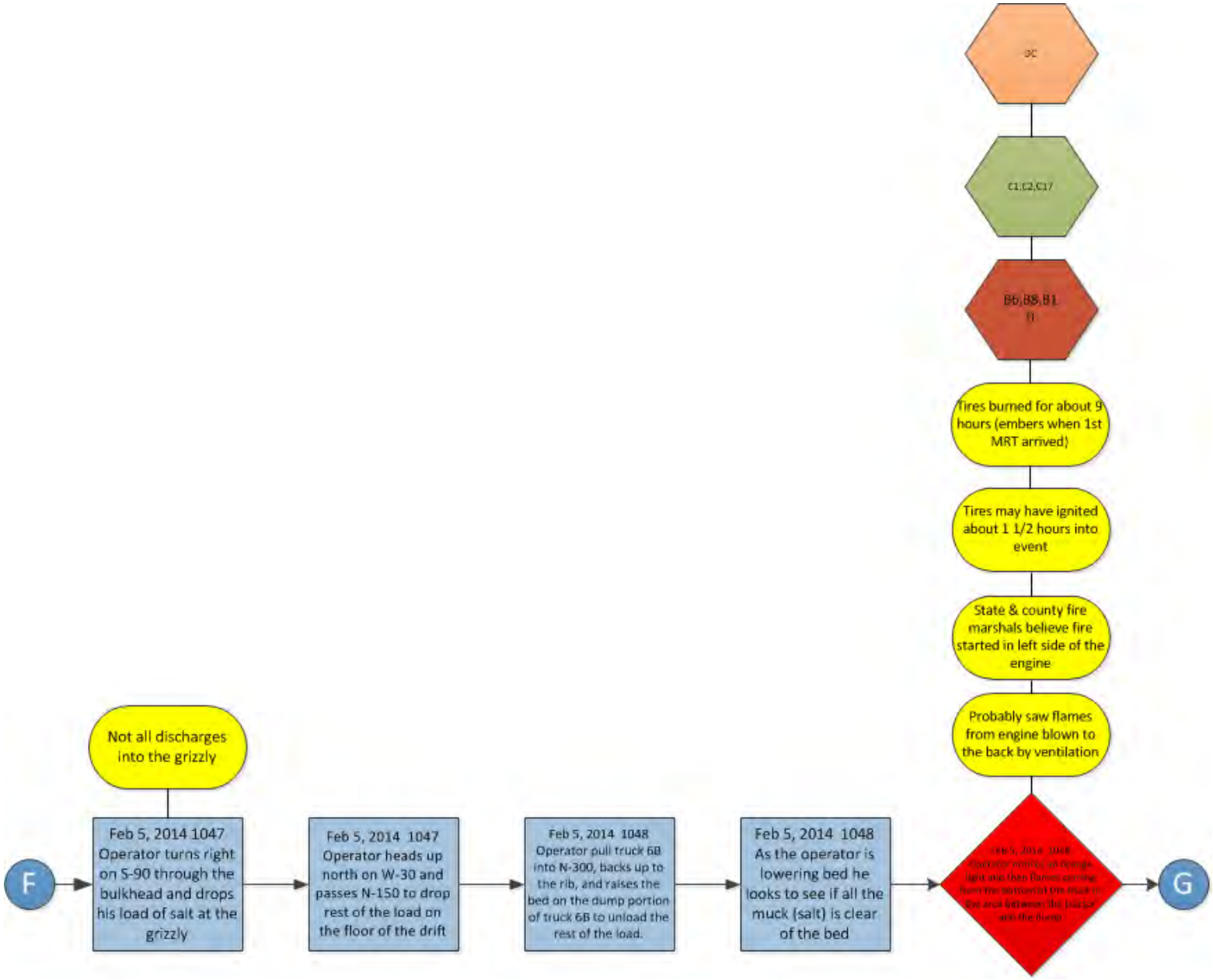

CC1, CC7, CC8, CC9, CC10
CC5, CC7, CC8, CC9
CC8, CC9
CC7, CC8, CC9
C37, C38, C39
C37, C38, C39
C37, C38, C39
C37, C38, C39
B27, B35, B36, B37, B38
B27, B35, B36, B37, B38
B27, B35, B36, B37, B38
B27, B35, B36, B37, B38
CC8
Truck is in continuous service through date of event, receives routine maint. & one rebuild
Many issues still being addressed/unresolved
Many issues still being addressed/unresolved (HS-45 and EMCBC reviews)
Many issues still being addressed/unresolved
CBFO in transition for 3 years (Mgr, Deputy Mgr
Many issues still being addressed/unresolved
Does not include an onboard fire suppression system (FSS)
Issues with quality and usability of work documents, compliance, post-maint testing, resources, CBFO oversight
Issues with EALs, categorization, classification
Issues with CBFO oversight, rigor, issue tracking, staff qualification
Change from PBI to PBI/ CPAF, change in ops manager
Issues with FR oversight, issues mgt, PMs
May, 1985 Eimco Salt Haul Truck 74-U-006B (Truck 6B) is purchased
March, 2012 DNFSB review of maintenance program
March 2012 HS-review of emergency management program
August 2012 EMCBC review of safety programs and oversight
October 2012 Change in contract to NWP
June 2013 EM-42 assist visit - maintenance management
A

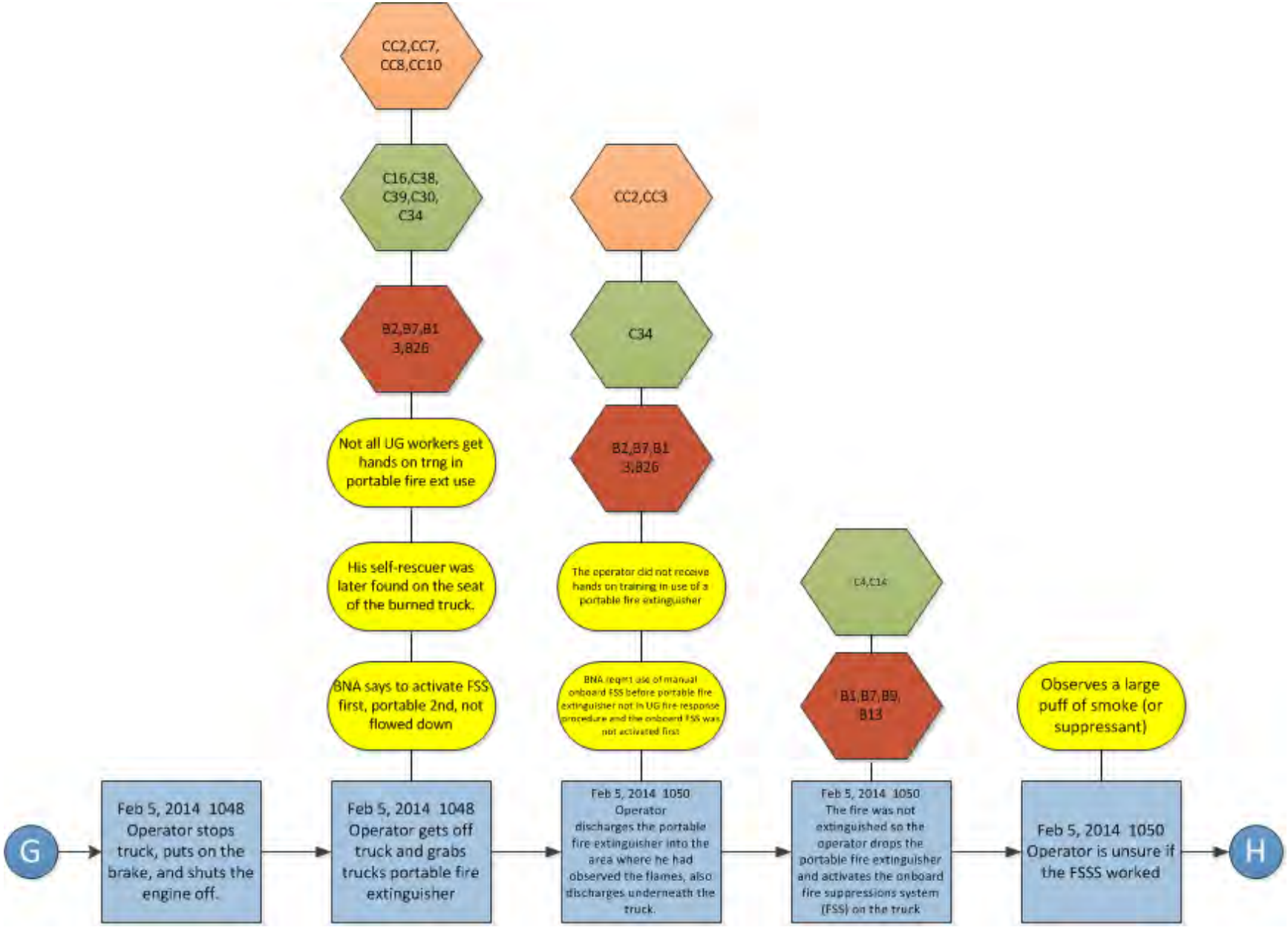

A
July 2013
EM-42 review of FR program
Issues with no implementing procedures, inadequate training, issue tracking & closure
Many issues still being addressed/unresolved
B27, B35, B36, B37, B38
C37, C38, C39
CC8, CC9
Oct 22, 2013
SW Safety performs PM on salt truck
Uses air instead of nitrogen (contrary to procedure) due to hazard concern
B27, B35, B36, B37, B38
C37, C38, C39
CC7, CC10
1985 – 2013
Truck 6B is in service, receives scheduled maintenance, at least 1 rebld
Aged equip (over 29 years)
Wash bay taken OOS (wash water is difficult to bring up)→ combustible build up occurs
Added auto FSS in ~1993, changed to manual 2003 due to inadvertent actuation, FSS is used once to put out electrical fire
Salt haul trucks are capable of using fire resistant hydraulic fluid but they are not used in these vehicles.
DSA/TSR LCO 3.3.7 does not prohibit a truck in this condition to be at the waste face
B1,B5,B8,B9, B11,B12,B20, B27,B38,B39
C3,C4,C17,C35,C37,C38, C39
CC1,CC2, CC6,CC7, CC8
Pre-1993
Automatic FSS is added to truck 6B
CC2
Oct-03
The automatic FSS is converted to manual activation
Decision to disable the automatic fire suppression system due to inadvertent actuation (engineering)
B26,B27, B35
C38,C39
CC1,CC2, CC7,CC8
B

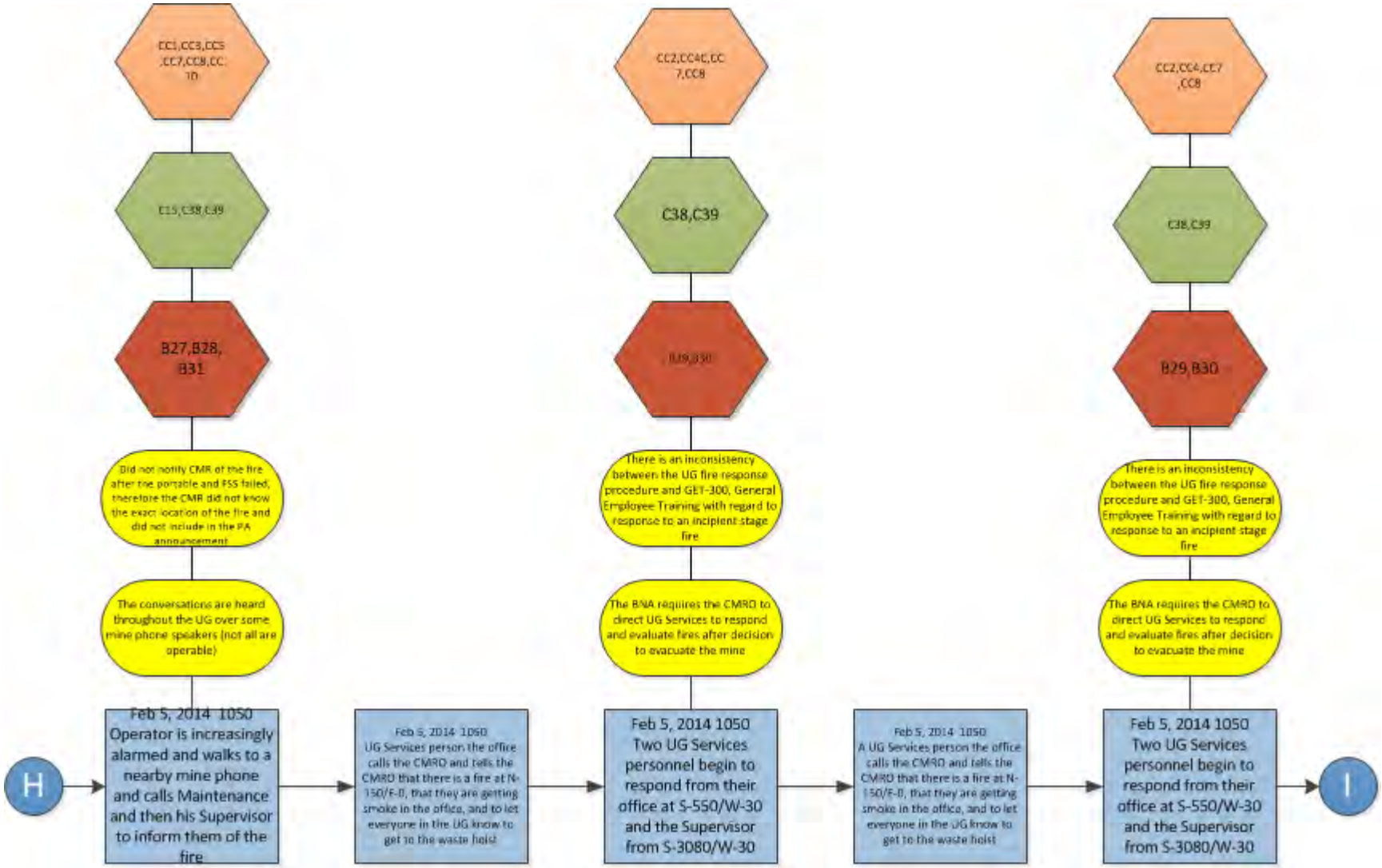

CC1,CC2,CC7
,CC8
C38,C39
B26,B27,
B35
Decision to disable the automatic fire suppression system due to inadvertent actuation (engineering).
Results satisfactory
Results satisfactory
B
Oct 2003
The automatic FSS is converted to manual activation
~2004
Truck 6B engine is rebuilt
Oct 23, 2013
Quarterly emissions tests was performed on truck 6B
Nov 26, 2013
100 hour preventative maintenance is performed on truck 6B
Dec 17, 2013
Batteries are replaced on truck 6B via an expedited work package
C


CC10
C18
Operator unaware that portable extinguisher has a gauge, supposed to inspect each shift
Daily pre-op checks don't indicate issues with combustible fluid bldup
Results satisfactory
Results satisfactory
Hoses are made onsite
Results satisfactory
C
Dec 21, 2013 500 hour preventative maintenance is performed on truck 6B
Jan 20, 2014 100 hour preventative maintenance is performed on truck 6B
Jan 21, 2014 Replaced hydraulic hose on truck 6B per expedited work package
Jan 24, 2014 Quarterly emissions tests was performed on truck 6B
Jan 24 – Feb 5, 2014 Truck 6B is in service hauling salt to grizzly
D


D
Feb 5, 2014 0000 Central Monitoring Room Operator(CMRO) puts ventilation system in maintenance bypass mode, filtration is enabled
Feb 5, 2014 0555 Salt hoist checks are completed
Feb 5, 2014 0816 The contact handled bay is configured for waste handling (WH) mode
Feb 5, 2014 The CMRO disables filtration for UG local processing unit (LPU) testing
Feb 5, 2014 0834 LPU testing is unsatisfactory (results require a manual shift to filtration upon a loss of power scenario)
E
E
Feb 5, 2014 0848 CMRO configures the shaft access area (SAA) and UG for contact handled (CH) waste handling
Feb 2, 2014 0859 CMRO configures the SAA and UG for remote handled (RH) waste handling
Feb 5, 2014 1045 CMRO configures the RH bay for waste handling
Feb 5, 2014 1046 Salt haul truck operator (Operator) is at Panel 8 in truck 6B and is loaded with salt (last load before lunch)
Feb 5, 2014 1046 Operator heads down W-170 in truck 6B towards the grizzly
F


DC
C1,C2,C17
B6,B8,B1
0
Tires burned for about 9 hours (embers when 1st MRT arrived)
Tires may have ignited about 1 1/2 hours into event
State & county fire marshals believe fire started in left side of the engine
Probably saw flames from engine blown to the back by ventilation
Not all discharges into the grizzly
F
Feb 5, 2014 1047 Operator turns right on S-90 through the bulkhead and drops his load of salt at the grizzly
Feb 5, 2014 1047 Operator heads up north on W-30 and passes N-150 to drop rest of the load on the floor of the drift
Feb 5, 2014 1048 Operator pull truck 6B into N-300, backs up to the rib, and raises the bed on the dump portion of truck 6B to unload the rest of the load.
Feb 5, 2014 1048 As the operator is lowering bed he looks to see if all the muck (salt) is clear of the bed
G


CC2,CC7, CC8,CC10
C16,C38, C39,C30, C34
B2,B7,B1 3,B26
Not all UG workers get hands on trng in portable fire ext use
His self-rescuer was later found on the seat of the burned truck.
BNA says to activate FSS first, portable 2nd, not flowed down
CC2,CC3
C34
B2,B7,B1 3,B26
The operator did not receive hands on training in use of a portable fire extinguisher
BNA reqmt use of manual onboard FSS before portable fire extinguisher not in UG fire response procedure and the onboard FSS was not activated first
C4,C14
B1,B7,B9, B13
Observes a large puff of smoke (or suppressant)
G
Feb 5, 2014 1048 Operator stops truck, puts on the brake, and shuts the engine off.
Feb 5, 2014 1048 Operator gets off truck and grabs trucks portable fire extinguisher
Feb 5, 2014 1050 Operator discharges the portable fire extinguisher into the area where he had observed the flames, also discharges underneath the truck.
Feb 5, 2014 1050 The fire was not extinguished so the operator drops the portable fire extinguisher and activates the onboard fire suppressions system (FSS) on the truck
Feb 5, 2014 1050 Operator is unsure if the FSSS worked
H


CC1,CC3,CC5,CC7,CC8,CC10
C15,C38,C39
B27,B28,B31
Did not notify CMR of the fire after the portable and FSS failed, therefore the CMR did not know the exact location of the fire and did not include in the PA announcement
The conversations are heard throughout the UG over some mine phone speakers (not all are operable)
H
Feb 5, 2014 1050 Operator is increasingly alarmed and walks to a nearby mine phone and calls Maintenance and then his Supervisor to inform them of the fire
Feb 5, 2014 1050 UG Services person the office calls the CMRO and tells the CMRO that there is a fire at N-150/F-0, that they are getting smoke in the office, and to let everyone in the UG know to get to the waste hoist
CC2,CC4,CC7,CC8
C38,C39
There is an inconsistency between the UG fire response procedure and GET-300, General Employee Training with regard to response to an incipient stage fire
The BNA requires the CMRO to direct UG Services to respond and evaluate fires after decision to evacuate the mine
Feb 5, 2014 1050 Two UG Services personnel begin to respond from their office at S-550/W-30 and the Supervisor from S-3080/W-30
Feb 5, 2014 1050 A UG Services person the office calls the CMRO and tells the CMRO that there is a fire at N-150/F-0, that they are getting smoke in the office, and to let everyone in the UG know to get to the waste hoist
CC2,CC4,CC7,CC8
C38,C39
B29,B30
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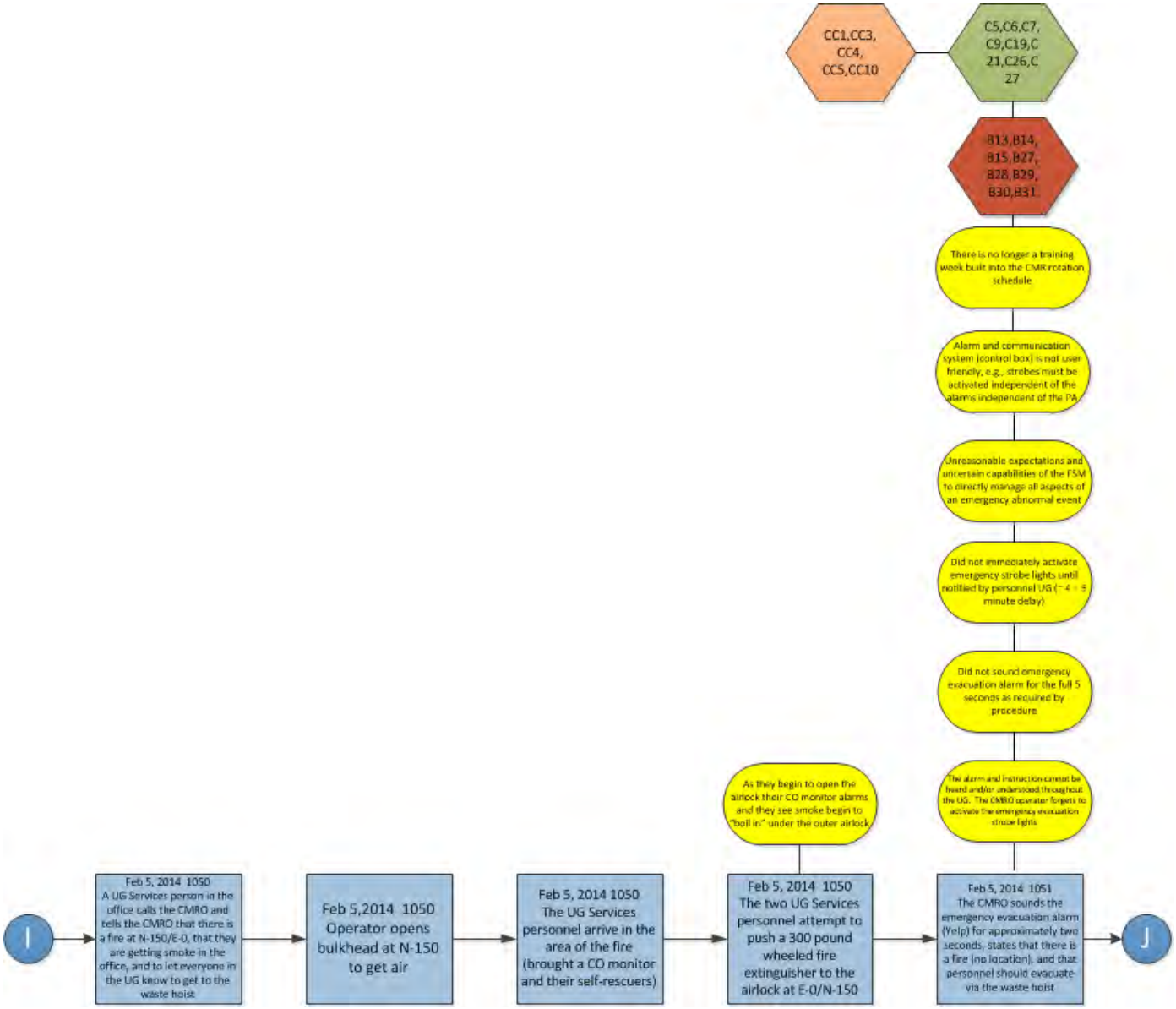
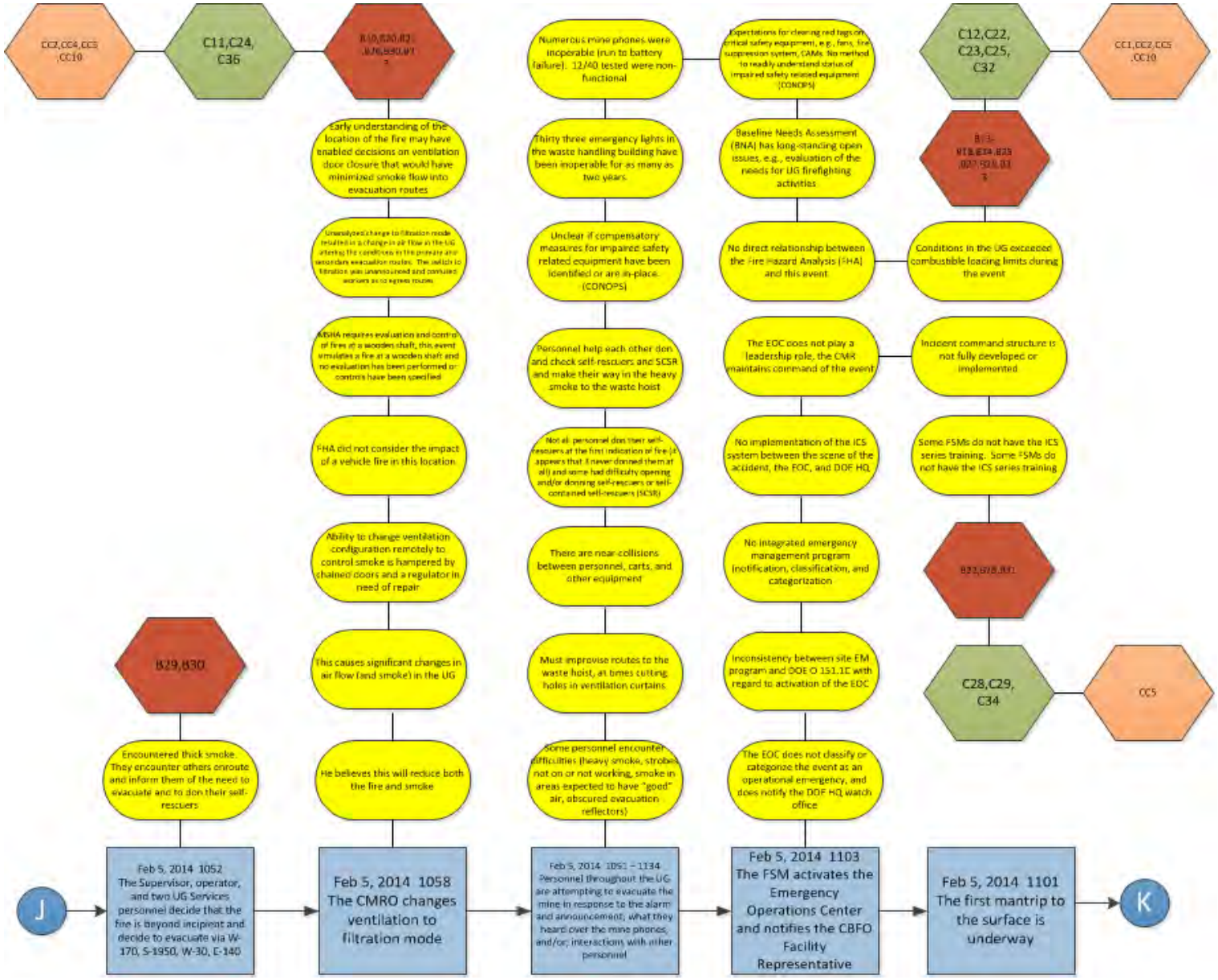
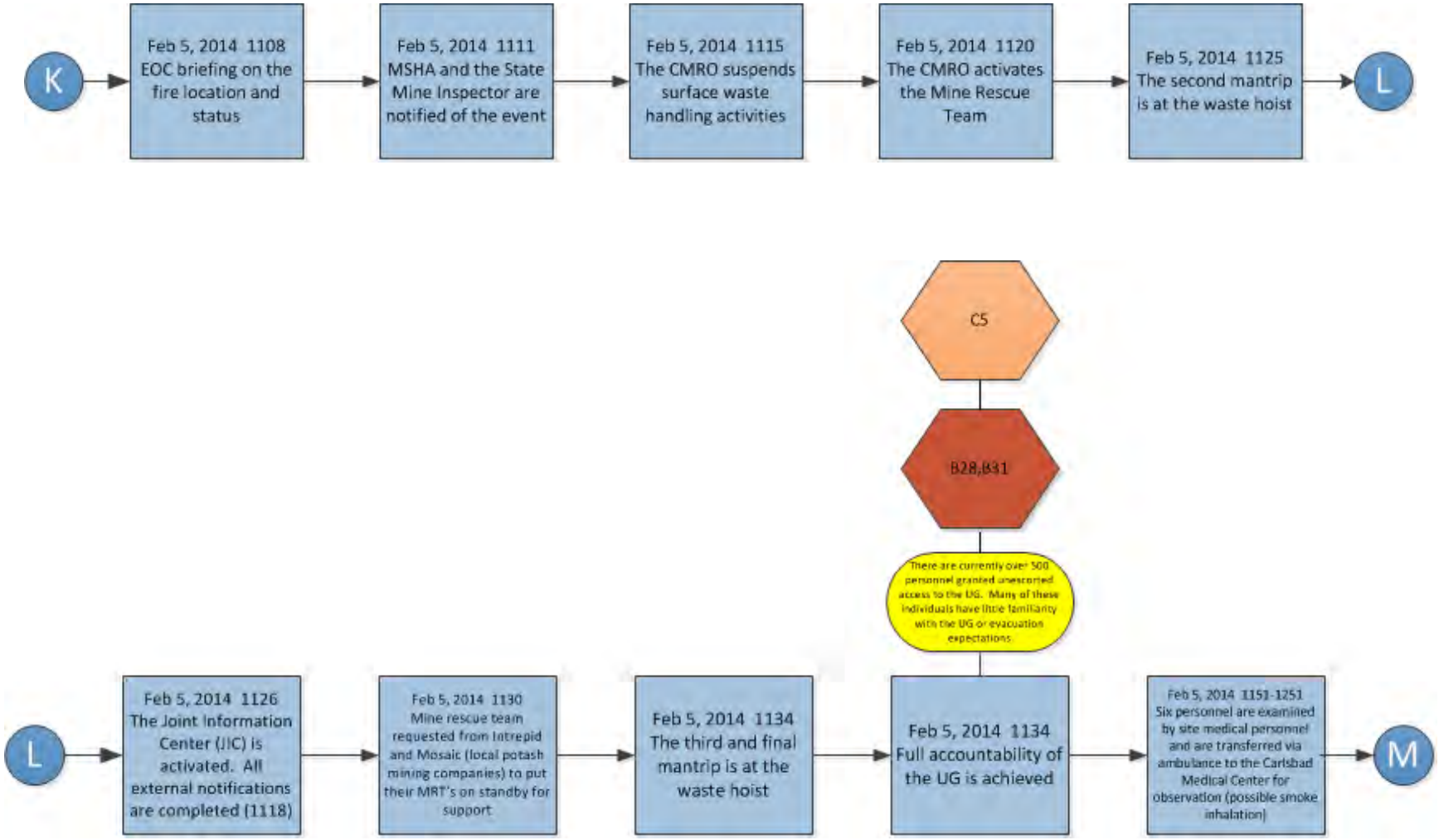
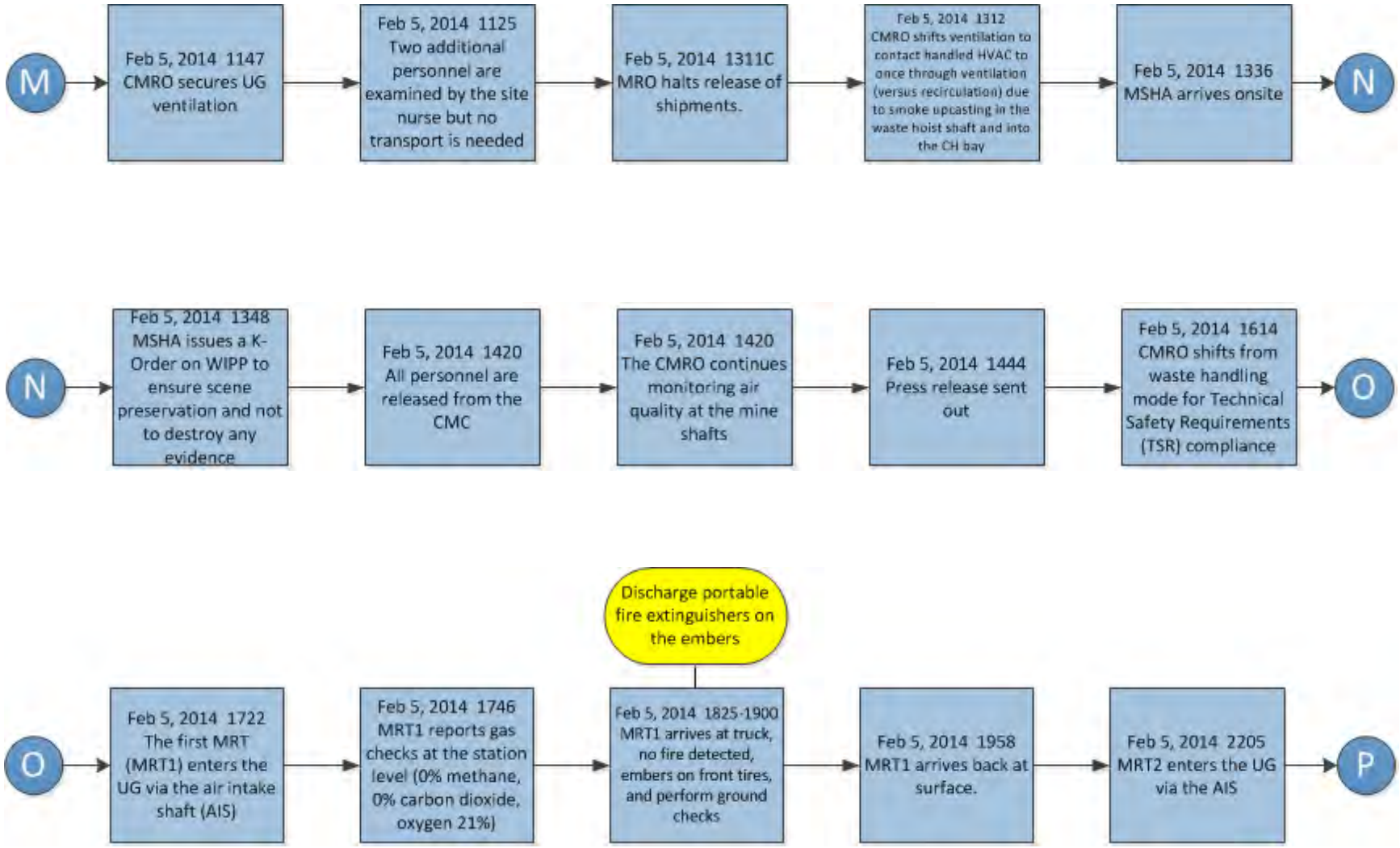
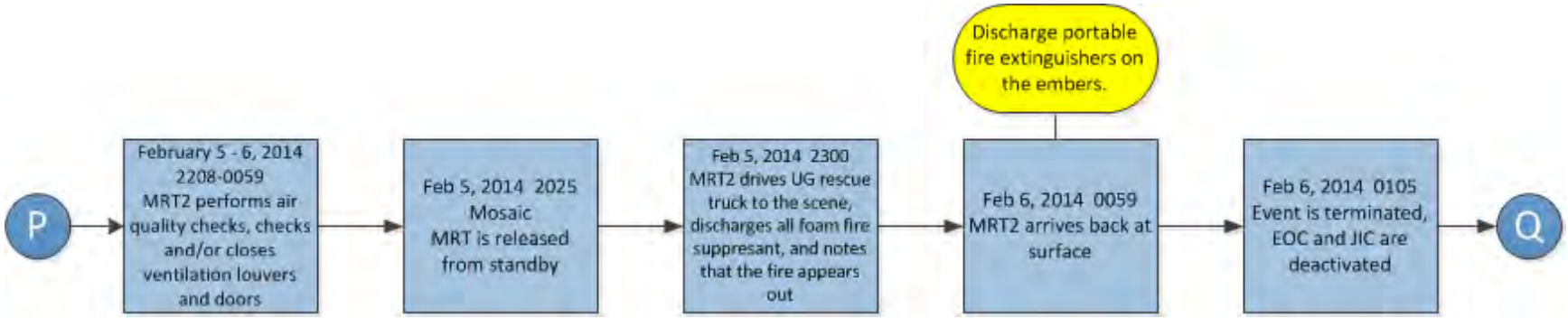
I

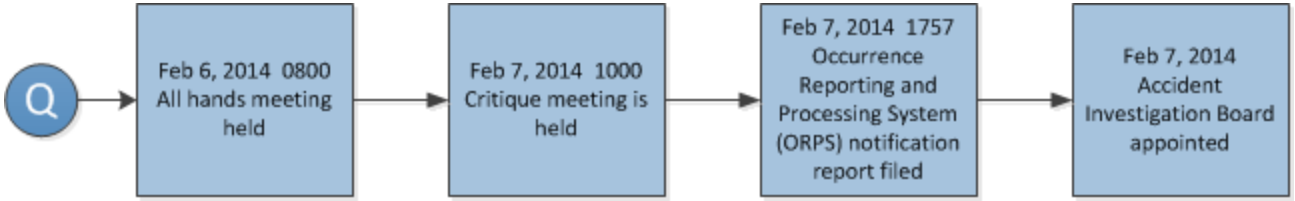

CC1,CC3, CC4, CC5,CC10
C5,C6,C7, C9,C19,C 21,C26,C 27
B13,B14, B15,B27, B28,B29, B30,B31
There is no longer a training week built into the CMR rotation schedule
Alarm and communication system (control box) is not user friendly, e.g., strobes must be activated independent of the alarms independent of the PA
Unreasonable expectations and uncertain capabilities of the FSM to directly manage all aspects of an emergency abnormal event
Did not immediately activate emergency strobe lights until notified by personnel UG (~ 4 - 5 minute delay)
Did not sound emergency evacuation alarm for the full 5 seconds as required by procedure
The alarm and instruction cannot be heard and/or understood throughout the UG. The CMRO operator forgets to activate the emergency evacuation strobe lights
As they begin to open the airlock their CO monitor alarms and they see smoke begin to "boil in" under the outer airlock
I
Feb 5, 2014 1050 A UG Services person in the office calls the CMRO and tells the CMRO that there is a fire at N-150/E-0, that they are getting smoke in the office, and to let everyone in the UG know to get to the waste hoist
Feb 5,2014 1050 Operator opens bulkhead at N-150 to get air
Feb 5, 2014 1050 The UG Services personnel arrive in the area of the fire (brought a CO monitor and their self-rescuers)
Feb 5, 2014 1050 The two UG Services personnel attempt to push a 300 pound wheeled fire extinguisher to the airlock at E-0/N-150
Feb 5, 2014 1051 The CMRO sounds the emergency evacuation alarm (Yelp) for approximately two seconds, states that there is a fire (no location), and that personnel should evacuate via the waste hoist
J


CC2,CC4,CC5
,CC10
C11,C24,
C36
Early understanding of the location of the fire may have enabled decisions on ventilation door closure that would have minimized smoke flow into evacuation routes
Unanalyzed change to filtration mode resulted in a change in air flow in the UG altering the conditions in the primary and secondary evacuation routes. The switch to filtration was unannounced and confused workers as to egress routes
MSHA requires evaluation and control of fires at a wooden shaft, this event simulates a fire at a wooden shaft and no evaluation has been performed or controls have been specified
FHA did not consider the impact of a vehicle fire in this location
Ability to change ventilation configuration remotely to control smoke is hampered by chained doors and a regulator in need of repair
This causes significant changes in air flow (and smoke) in the UG
He believes this will reduce both the fire and smoke
B29,B30
Encountered thick smoke. They encounter others enroute and inform them of the need to evacuate and to don their self-rescuers
J
Feb 5, 2014 1052
The Supervisor, operator, and two UG Services personnel decide that the fire is beyond incipient and decide to evacuate via W-170, S-1950, W-30, E-140
Feb 5, 2014 1058
The CMRO changes ventilation to filtration mode
Numerous mine phones were inoperable (run to battery failure). 12/40 tested were non-functional
Thirty three emergency lights in the waste handling building have been inoperable for as many as two years
Unclear if compensatory measures for impaired safety related equipment have been identified or are in-place. (CONOPS)
Personnel help each other don and check self-rescuers and SCSR and make their way in the heavy smoke to the waste hoist
Not all personnel donned their self-rescuers at the first indication of fire (it appears that it never donned them at all) and some had difficulty opening and/or donning self-rescuers or self-contained self-rescuers (SCSR)
There are near-collisions between personnel, carts, and other equipment
Must improvise routes to the waste hoist, at times cutting holes in ventilation curtains
Some personnel encounter difficulties (heavy smoke, strobes not on or not working, smoke in areas expected to have "good" air, obscured evacuation reflectors)
Feb 5, 2014 1051 – 1134
Personnel throughout the UG are attempting to evacuate the mine in response to the alarm and announcement; what they heard over the mine phones, and/or; interactions with other personnel
Expectations for clearing red tags on critical safety equipment, e.g., fans, fire suppression system, CAMs. No method to readily understand status of impaired safety related equipment (CONOPS)
Baseline Needs Assessment (BNA) has long-standing open issues, e.g., evaluation of the needs for UG firefighting activities
No direct relationship between the Fire Hazard Analysis (FHA) and this event
The EOC does not play a leadership role, the CMR maintains command of the event
No implementation of the ICS system between the scene of the accident, the EOC, and DOE HQ
No integrated emergency management program (notification, classification, and categorization
Inconsistency between site EM program and DOE O 151.1C with regard to activation of the EOC
The EOC does not classify or categorize the event as an operational emergency, and does notify the DOE HQ watch office
Feb 5, 2014 1103
The FSM activates the Emergency Operations Center and notifies the CBFO Facility Representative
C12,C22,
C23,C25,
C32
CC1,CC2,CC5
,CC10
Conditions in the UG exceeded combustible loading limits during the event
Incident command structure is not fully developed or implemented
Some FSMs do not have the ICS series training. Some FSMs do not have the ICS series training
B22,B28,B31
C28,C29,
C34
CC5
Feb 5, 2014 1101
The first mantrip to the surface is underway
K


K
Feb 5, 2014 1108 EOC briefing on the fire location and status
Feb 5, 2014 1111 MSHA and the State Mine Inspector are notified of the event
Feb 5, 2014 1115 The CMRO suspends surface waste handling activities
Feb 5, 2014 1120 The CMRO activates the Mine Rescue Team
Feb 5, 2014 1125 The second mantrip is at the waste hoist
L
C5
B28,B31
There are currently over 500 personnel granted unescorted access to the UG. Many of these individuals have little familiarity with the UG or evacuation expectations
L
Feb 5, 2014 1126 The Joint Information Center (JIC) is activated. All external notifications are completed (1118)
Feb 5, 2014 1130 Mine rescue team requested from Intrepid and Mosaic (local potash mining companies) to put their MRT's on standby for support
Feb 5, 2014 1134 The third and final mantrip is at the waste hoist
Feb 5, 2014 1134 Full accountability of the UG is achieved
Feb 5, 2014 1151-1251 Six personnel are examined by site medical personnel and are transferred via ambulance to the Carlsbad Medical Center for observation (possible smoke inhalation)
M


M
Feb 5, 2014 1147 CMRO secures UG ventilation
Feb 5, 2014 1125 Two additional personnel are examined by the site nurse but no transport is needed
Feb 5, 2014 1311C MRO halts release of shipments.
Feb 5, 2014 1312 CMRO shifts ventilation to contact handled HVAC to once through ventilation (versus recirculation) due to smoke upcasting in the waste hoist shaft and into the CH bay
Feb 5, 2014 1336 MSHA arrives onsite
N
N
Feb 5, 2014 1348 MSHA issues a K-Order on WIPP to ensure scene preservation and not to destroy any evidence
Feb 5, 2014 1420 All personnel are released from the CMC
Feb 5, 2014 1420 The CMRO continues monitoring air quality at the mine shafts
Feb 5, 2014 1444 Press release sent out
Feb 5, 2014 1614 CMRO shifts from waste handling mode for Technical Safety Requirements (TSR) compliance
O
Discharge portable fire extinguishers on the embers
O
Feb 5, 2014 1722 The first MRT (MRT1) enters the UG via the air intake shaft (AIS)
Feb 5, 2014 1746 MRT1 reports gas checks at the station level (0% methane, 0% carbon dioxide, oxygen 21%)
Feb 5, 2014 1825-1900 MRT1 arrives at truck, no fire detected, embers on front tires, and perform ground checks
Feb 5, 2014 1958 MRT1 arrives back at surface.
Feb 5, 2014 2205 MRT2 enters the UG via the AIS
P


P
February 5 - 6, 2014
2208-0059
MRT2 performs air quality checks, checks and/or closes ventilation louvers and doors
Feb 5, 2014 2025
Mosaic
MRT is released from standby
Feb 5, 2014 2300
MRT2 drives UG rescue truck to the scene, discharges all foam fire suppresant, and notes that the fire appears out
Discharge portable fire extinguishers on the embers.
Feb 6, 2014 0059
MRT2 arrives back at surface
Feb 6, 2014 0105
Event is terminated, EOC and JIC are deactivated
Q


Q
Feb 6, 2014 0800
All hands meeting held
Feb 7, 2014 1000
Critique meeting is held
Feb 7, 2014 1757
Occurrence Reporting and Processing System (ORPS) notification report filed
Feb 7, 2014
Accident Investigation Board appointed

# Appendix F. **Report from Fire Investigators**

**Fire Investigation Report**
**Waste Isolation Pilot Project**
**Investigator: Robert Brader**

This report is written as a supplemental report for the investigation team that is reviewing a fire that occurred on February 5, 2014 underground at the Waste Isolation Pilot Project (WIPP). On February 19, 2014 I, along with William Farmer of the New Mexico State Fire Marshal's Office, met with representatives from the Department of Energy (DOE), the URS Corporation, and the Defense Nuclear Safety Board (DNFSB) at the DOE Carlsbad Field Office aka the Skeen Whitlock Building. The purpose of the meeting was to review photos, drawings, and information provided from team members to assist in identifying the origin and cause of the fire. Due to other complications the fire scene could not be directly accessed.

**Vehicle Description:**

This fire occurred in a mining vehicle identified as an Eimco 895D T15 Haul Truck. Photos of the truck show it to be an industrial mining truck with a dump bed to the rear, an open cab operator section in the center and the engine compartment to the front. The vehicle has four tires. The rear two are located under the center of the dump bed and the front two are located behind the operators section placing them behind the operator section and the engine compartment. The vehicle has a diesel motor, hydraulic over mechanical brakes, hydraulic dump systems, various electrical systems, and an onboard manually activated dry chemical fire fighting system. Ignitable liquids include Diesel fuel, Hydraulic Fluid, Engine Oil, and lubricating Grease. Major fuel packages include ignitable liquids, tires, and the seat cushion. Readily identifiable potential ignition sources include hot surfaces especially engine exhaust components, electrical wiring, and friction heat from mechanical component failure.

**Location of the Fire:**

This fire occurred underground at a "T" intersection of three tunnels. Photographs drawings and team description of the location were reviewed. This intersection is created where a major ventilation tunnel coming from the air intake shaft intersects what I will term as the "uphill" tunnel, leading to the salt handling shaft, where the haul truck was dumping salt and the "downhill" tunnel leading to the mining area, near the exhaust shaft ,where the truck was bringing salt from.

**Airflow:**

It is my impression that airflow played a significant role in fire propagation. Based on information provided by the team it appears that in unobstructed flow air moves through the ventilation tunnel from the intake shaft and then diverges in opposite directions following both the "uphill" and "downhill" tunnels. Airflow at this point during normal ventilation was described as over 400,000 CFM in the "downhill" tunnel and over 100,000 CFM in the "uphill". They also stated that during the fire the airflow flow was reduced to the 60,000 CFM range.

**History of the Fire:**

The vehicle was reported to have been in use for approximately 29 years at this location. On the day of the fire the vehicle is reported to have been in continuous use for approximately 4 hours. The operator described the incident as follows. He was dumping a load of salt in the "uphill" tunnel when he noticed a glow reflecting off the bottom of the raised dump bed. He then lowered the bed and drove forward to the intersection of the tunnels and exited the vehicle to identify the glow he had seen. He parked the vehicle in the intersection with the front toward the "downhill" tunnel, the rear toward the "uphill tunnel, the right hand side toward the ventilation tunnel, and the left hand side toward the salt rib. He

located a hole in the frame of the vehicle near the mid-point of the right hand side of the vehicle. He discharged a hand held dry-chemical extinguisher into this hole. When this did not extinguish what he had determined to be a fire he activated the onboard dry chemical extinguisher which did not extinguish the fire. He then abandoned the vehicle and evacuated. The vehicle was allowed to burn unimpeded for several hours before rescue crews re-entered the mine and extinguished the then smoldering fire.

**Movement and Intensity Indicators:**

From a broad overview photographs show a large truncated cone pattern on the salt rib starting near the engine compartment and extending up and out toward the "downhill" shaft. This pattern is leaned over from the vertical toward the "downhill" horizontal. This is consistent with an air driven fire coming from the front portion of the vehicle and being pushed in the "downhill" direction and back against the opposing rib. This pattern would have been created during the time of high airflow and as such occurred early in the fire prior to the change in ventilation. Also seen from broad overview photos is that the damage clearly is more severe in the front of the vehicle and progresses to relatively undamaged at the rear of the vehicle. The rear tires are intact; the front tires are burned away. This is consistent with a fire moving from the front of the vehicle to the rear of the vehicle. Close up photos show movement and intensity patterns and degree of damage leading from the rear of the truck back toward the engine compartment. This is consistent with the operator discovering the fire below and forward of the operators area. A hydraulic accumulator that was located in the operator's area on the floor was found by the team to have BELVED. Since this was not described by the operator and would have clearly impacted him had it occurred during operation this is consistent with having occurred later in the fire. This would have added well heated ignitable liquids to the operator's area aiding in fire propagation to that area. Photos of the hole where the operator discharged the hand extinguisher show burn marks around the hole that appear to be air driven toward the rear of the vehicle. This is consistent with the theory of fire propagation offered later in this report and is not primarily indicative of the area of origin. Photos of the exterior and interior of the engine compartment continue to support the fire originating in the engine compartment. Here degree of damage and movement patterns support the fire having started on the left side of the engine compartment down low and wrapping up and over the engine and to the right side. A classic "V" pattern on the front of the truck leads to a point low down near the base of the engine compartment and appears to wrap around from the left side. Photos of the left side of the vehicle and left interior engine compartment were not available due to safety concerns about potential collapse of the heat impacted salt rib.

**Theory of Materials First Ignited, Area of Origin, and Fire Propagation:**

It is impossible to be dogmatic about the origin and cause of this fire given the limitations of evidence, my inability to directly examine the vehicle, and the inability of those team members who had accessed the vehicle to fully examine and photograph it. That being said it is possible to make reasonable inferences and develop a most likely scenario of fire propagation.

The evidence clearly supports this fire starting low down on the driver's side of the vehicle. The major fuel package in this area would be the ignitable liquids.

While any of the ignitable liquids could have been the material first ignited, including an accumulated mixture in the belly pan, I believe the most plausible would be discharge of hydraulic fluid under pressure on to exhaust components. It should be noted that the exhaust transits the area of origin to the catalytic converter located just on the outside of the engine compartment. Anecdotal evidence from a brief internet search of known failures of this type of equipment and brief interviews with local acquaintances who have operated or repaired similar mining equipment indicates that hydraulic failure is not uncommon and hydraulic fluid contacting a hot surface may be a leading cause of these types of

fires in local mines. I was told that any such fire lasting less than 30 minutes would not be reported to MSHA and thus account for limited published data on this type of fire.

I believe that this fire ignited during the normal operation of the truck. The fire ignited and then continued burning low down in the belly pan and was eventually being fed by one or several ignitable liquids as other lines in the area failed. The glow seen by the operator was the fire being reflected down the belly pan and up on to the bottom of the dump bed. The location where this happened would have been up the "uphill" shaft with the vehicle pointed back towards the intersection. This would have had the ventilation air blowing directly in to the front of the truck. This airflow would have been enhanced by both the engine cooling fan forcing air from the front of the vehicle toward the rear and the forward motion of the vehicle. This would account for the fire being briefly pushed toward the rear and out the hole in the right side. This would produce the patterns previously discussed and further support the finding that this fire started in the forward part of the engine compartment.

When the operator lowered the bed and moved forward he did not experience any recognizable equipment failure. This helps preclude an electrical short, mechanical heating of a failed component such as wheel bearings, and catastrophic engine or transmission failure. It would be expected that the hydraulic system would continue to work even in the face of a leak for some period of time until the fluid reservoir ran low.

Once the driver moved the vehicle the airflow would dramatically change. As the truck entered the intersection the flow would change from what can be described as a head wind to a right front angled side wind, to a broad side wind to a right rear angled side wind and eventually to a tail wind. In the location where the truck was stopped it appears it would have had a right rear angled side wind. This is consistent with the movement and intensity patterns seen. In this location the airflow would have pushed the fire into the left front of the engine compartment and held it away from spreading to the rest of the truck. Furthermore once the vehicle was shut off the cooling fan would no longer be pushing airflow to the rear of the vehicle. The onboard fire suppression system appears to be designed and intended to discharge into the engine compartment, Photos of the engine compartment do not show significant amounts of extinguisher powder leading to the apparent conclusion that the system did not perform as designed. Witness information from outside the mine indicates that the smoke column exiting the exhaust shaft changed to a heavy black smoke that smelled like burning rubber after the air flow was reduced. This is consistent with the fire no longer being air driven to the front of the vehicle and propagating toward the rear of the vehicle and thus igniting the front tires. The heat from the tires burning impacting the diesel and hydraulic tanks near the operator's area accounts for any remaining fluids that had not leaked into the engine compartment being vaporized.

**Conclusion:**

In conclusion this was most likely an accidental fire resulting from an unidentified failure that allowed ignition of ignitable fluids in the front right engine compartment that the progressed rearward to the operator's area and the front tires.

# EXHIBIT 21

# AUDIT REPORT

Audit of NRC's Oversight of Active Component Aging

OIG-14-A-02 –October 28, 2013

All publicly available OIG reports (including this report) are accessible through NRC's Web site at:
http:/www.nrc.gov/reading-rm/doc-collections/insp-gen/

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICE OF THE
INSPECTOR GENERAL

October 28, 2013

MEMORANDUM TO: Mark A. Satorius
Executive Director for Operations

FROM: Stephen D. Dingbaum ***/RA/***
Assistant Inspector General for Audits

SUBJECT: AUDIT OF NRC'S OVERSIGHT OF ACTIVE COMPONENT AGING (OIG-14-A-02)

Attached is the Office of the Inspector General's (OIG) audit report titled *Audit of NRC's Oversight of Active Component Aging.*

The report presents the results of the subject audit. The agency provided comments to the report on September 27, 2013. The agency's comments have been incorporated into the report at Appendix B.

Please provide information on actions taken or planned on each of the recommendations within 30 days of the date of this memorandum. Actions taken or planned are subject to OIG followup as stated in Management Directive 6.1.

We appreciate the cooperation extended to us by members of your staff during the audit. If you have any questions or comments about our report, please contact me at 415-5915 or R.K. Wild, Team Leader, Nuclear Reactor Safety Audits Team, at 415-5948.

Attachment: As stated

# EXECUTIVE SUMMARY

## BACKGROUND

The Atomic Energy Act and Nuclear Regulatory Commission (NRC) regulations limit commercial nuclear power reactor licenses to an initial 40 years. Due to this selected period, some components may have been engineered on the basis of an expected 40-year service life. Components degraded due to aging have caused reactor shutdowns, failure of safety-related equipment, and reduction in the safety margin of operating nuclear power plants. Therefore, effective and proactive management of aging of components is a key element for safe and reliable nuclear power plant operation.

NRC has established commercial nuclear power reactor industry requirements that exclude some components—referred to as active components—from a license renewal aging management review. Active components are those that perform their intended functions with moving parts or a change in state. Examples of active components include power supplies, motors, diesel generators, cooling fans, batteries, relays, and switches. According to NRC, active components are not subject to review as part of NRC's review of license renewal applications because of the existing regulatory process and existing licensee programs and activities.

The NRC Office of Nuclear Reactor Regulation and the regional offices provide regulatory oversight of industry's active component aging activities. NRC addresses aging active component issues through a number of different regulations and guidance, to include Title 10 Code of Federal Regulations (CFR), Part 50.65*, Requirements for monitoring the effectiveness of maintenance at nuclear power plants* (the Maintenance Rule, as amended)*,* 10 CFR Part 50, Appendix B*, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,* and 10 CFR 50.36*, Technical specifications.*

## OBJECTIVE

The objective of this audit was to determine if NRC is providing effective oversight of industry's aging component programs.

## RESULTS IN BRIEF

### Oversight of Active Component Aging Could Be Improved

Oversight of licensees' activities, including active component aging, should be structured and coordinated. However, NRC's approach for oversight of licensees' management of active component aging is not focused or coordinated. This has occurred because NRC has not conducted a systematic evaluation of program needs for overseeing licensees' aging management for active components since the establishment of the Reactor Oversight Process (ROP) in 2000, and does not have mechanisms for systematic and continual monitoring, collecting, and trending of age-related data for active components. Consequently, NRC cannot be fully assured that it is effectively overseeing licensees' management of aging active components.

## RECOMMENDATIONS

This report makes two recommendations to improve the agency's oversight of aging active component activities.

## AGENCY COMMENTS

During an August 20, 2013, exit conference and an August 26, 2013, staff meeting, agency provided informal comments, which OIG subsequently incorporated into the draft report as appropriate.

On September 27, 2013, NRC provided formal comments to the draft report. The agency's stated that its oversight of active component aging issues is being effectively dealt with under existing oversight procedures, and that its existing ROP provides the framework for ensuring that aging issues with the potential to impact safety are addressed in a timely manner.

OIG's central message is that because the agency uses regulations and inspections procedures for oversight of aging issues that predate the 2000

implementation of ROP and has not since evaluated whether this regulations and procedures still function as intended, the agency is not able to determine the effectiveness of aging component oversight.

Appendix A contains the audit Objective, Scope, and Methodology; Appendix B contains a copy of the agency's formal comments; and Appendix C contains OIG's analysis of the agency's formal comments.

## ABBREVIATIONS AND ACRONYMS

| | |
|---|---|
| CFR | Code of Federal Regulations |
| IOEB | Operating Experience Branch |
| INL | Idaho National Laboratory |
| NRC | Nuclear Regulatory Commission |
| NRR | Office of Nuclear Reactor Regulation |
| OIG | Office of the Inspector General |
| ROP | Reactor Oversight Process |

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# I. BACKGROUND

The Atomic Energy Act and Nuclear Regulatory Commission (NRC) regulations limit commercial nuclear power reactor licenses to an initial 40 years. Due to this selected period, some components may have been engineered on the basis of an expected 40-year service life. However, components that have aged during the 40-year period can have impacts on both the safety and the performance of nuclear power plants. Components degraded due to aging have caused reactor shutdowns, failure of safety-related equipment, and reduction in the safety margin of operating nuclear power plants. Therefore, effective and proactive management of aging of components is a key element for safe and reliable nuclear power plant operation.

Aging is defined as a general process in which characteristics of components[1] gradually change with time or use. Some examples of aging mechanisms include wear, fatigue, erosion, microbiological fouling, embrittlement, and chemical or biological reactions, or combinations of these processes. Proactive aging management means management of the aging of components that is implemented with foresight and anticipation throughout the component's lifetime.[2]

NRC has established commercial nuclear power reactor industry requirements that exclude some components—referred to as active components—from a license renewal aging management review.[3] Active components are those that perform their intended functions with moving parts or a change in state. Examples of active components include power supplies, motors, diesel generators, cooling fans, batteries, relays, and switches. According to NRC, active components are not subject to review as part of NRC's review of license renewal applications because of the existing regulatory process and existing licensee programs and activities.

[1] For the purposes of this report, the Office of the Inspector General (OIG) uses the term components in place of safety-related structures, systems, and components.
[2] *Proactive Management of Ageing for Nuclear Power Plants*, International Atomic Energy Agency, Vienna, 2009.
[3] Commercial power reactor licenses can and have been renewed beyond 40 years. As part of license renewal, plants undergo an aging management review by the NRC that includes passive components. Passive components are components that perform an intended function (as described in Title 10 Code of Federal Regulations (CFR) Part 54.4) without moving parts or without a change in configuration or properties and include the reactor vessel, steam generators, and ventilation ducts.

## Oversight Responsibility for Active Component Aging Activities

The NRC Office of Nuclear Reactor Regulation (NRR) and the regional offices provide regulatory oversight of industry's active component aging activities. Agency officials stated that NRC oversight of reactor licensees is conducted within the Reactor Oversight Process (ROP) framework. The ROP is the agency's program to inspect, measure, and assess the safety performance of commercial nuclear power plants and to respond to any decline in performance. OIG auditors did not review the entire ROP framework. OIG focused on NRC's active component aging-related oversight activities both within and outside the ROP framework.

Primarily, two branches in the NRR Division of Inspections and Regional Support—as well as the Divisions of Reactor Projects in NRC regional offices—have responsibility for regulatory oversight of licensee programs, which would include licensee management of active component aging. Within NRR, Reactor Inspection Branch responsibilities include providing programmatic leadership and support for activities associated with inspecting and assessing licensee performance at commercial nuclear power plants. This includes providing the necessary infrastructure for the inspection program in coordination with the regional offices, and supporting enhanced inspection teams and the ROP. The Operating Experience Branch (IOEB) collects, evaluates, and communicates information that may have caused domestic and international reactor events and provides lessons learned from those events to headquarters, the regions, and licensees. While not responsible for oversight, the Office of Nuclear Regulatory Research provides support that includes technical advice, technical tools, and information for identifying and resolving safety issues, including for aging phenomena.

## Regulations Applicable to Active Component Aging Oversight

No Federal law or regulation that pertains to NRC specifically provides for the oversight of aging active components. However, NRC inspectors have used the following regulations to support a basis for age-related inspection findings and violations:

- 10 CFR Part 50.65*, Requirements for monitoring the effectiveness of maintenance at nuclear power plants* (the Maintenance Rule, as amended) was issued on July 10, 1991.

The Maintenance Rule requires, in part, that licensees:

> ...shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components […] are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. […] Monitoring […] is not required where it has been demonstrated that the performance or condition of a structure, system, or component is being effectively controlled through the performance of appropriate preventive maintenance… .

- 10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants* (10 CFR Part 50, Appendix B), requires licensees to maintain a quality assurance program for the design, fabrication, construction, and testing of the structures, systems, and components of the facility.

- 10 CFR Part 50.36, *Technical specifications*, requires licensees to maintain administrative controls, including procedures for maintenance, to assure operation of the facility in a safe manner. Inspectors sometimes cite licensees for inadequate procedures that led to age-related degradation under this regulation.

Additionally, there are ROP and other inspection procedures, as well as industry operating experience that offer NRC inspectors and staff flexibility when used for active component aging oversight. Some, but not all, relevant inspection procedures, reports, and studies are listed in Appendix A of this report.

# II. OBJECTIVE

The objective of this audit was to determine if NRC is providing effective oversight of industry's aging component programs.

Appendix A of this report contains information on the audit scope and methodology.

# III. FINDING

## OVERSIGHT OF ACTIVE COMPONENT AGING COULD BE IMPROVED

Oversight of licensees' activities, including active component aging, should be structured and coordinated. However, NRC's approach for oversight of licensees' management of active component aging is not focused or coordinated. This has occurred because NRC has not conducted a systematic evaluation of program needs for overseeing licensees' aging management for active components since the establishment of ROP in 2000, and does not have mechanisms for systematic and continual monitoring, collecting, and trending of age-related data for active components. Consequently, NRC cannot be fully assured that it is effectively overseeing licensees' management of aging active components.

### Structured Oversight

Oversight of licensee's activities should be structured, coordinated, and based on the best available knowledge from research and operational experience. Commonly accepted, formal approaches to planning of Government programs include establishing an overall strategy and goals, establishing methodologies for setting priorities, identifying program-specific performance metrics, and managing resources. Additionally, *NRC Principles of Good Regulation* require that NRC manages and administers its regulatory activities cooperatively and efficiently, that regulatory decisions should be made without undue delay, and that regulations should be based on the best available knowledge from research and operational experience.

Furthermore, current and former NRC Commissioners, senior management officials, and managers have expectations that the agency is providing effective oversight of industry's active component aging programs and have repeatedly emphasized the importance of aging oversight. During the 2013 NRC Regulatory Information Conference, the current NRC Chairman noted that despite an established rigorous program for aging management, NRC and industry must be prepared to contend with unknowns. Former NRC Commissioners, senior management officials, and managers have also expressed the importance of strong aging management programs, including those for active components, and one senior manager stated that NRC currently has programs in place to monitor aging of active components. Agency managers have also emphasized the importance of oversight for active component aging activities. However, current and former managers having expectations for effective oversight of industry's aging plants generally did not make a distinction between active components and those passive components covered by license renewal aging management reviews.

**Oversight Structure for Aging Active Component Activities Is Not Focused or Coordinated**

NRC's approach for oversight of licensees' management of active component aging is not focused or coordinated. This approach includes staff-initiated projects and inspection activities using regulations to cite licensees for age-related degradation of active components that are not specific to aging. This challenge is compounded by agency senior managers who are not aware of these uncoordinated activities.

Staff-Initiated Projects

NRC program offices in headquarters and the regions have undertaken staff-initiated projects to evaluate age-related active component failures. Specifically, the staff-initiated projects have been for data collection, analysis, and inspection.

Independent of any specific management direction, NRC staff have initiated efforts to obtain data for addressing the subject of aging active

components because operating experience data is not routinely evaluated for information pertaining to the aging of active components. In 2012, NRR's Operating Experience Branch (IOEB) published the *IOEB Component Aging Study 2007-2011 — Insights from Inspection Findings and Reportable Events*, July 24, 2012 (*IOEB Study*).[4] The *IOEB Study* focused on safety-related and important-to-safety active component failures attributed to age-related degradation. In part, the *IOEB Study* noted that the number of occurrences involving age-related active component failures has increased since 2009. The *IOEB Study* concluded that some licensees do not have effective life-cycle preventive maintenance programs for some components where industry and vendor experience has suggested this is necessary. Furthermore, the *IOEB Study* concluded that NRC oversight programs may not be focused on aging management of active components and these programs could be better prepared to deal effectively with an industry that potentially is experiencing notable occurrences of age-related component failures.

The results of the *IOEB Study* prompted other NRC offices to generate active component age-related projects, again, without top agency management oversight and without any coordination. The Office of Nuclear Regulatory Research reviewed available active component age-related data[5] contained in the Idaho National Laboratory (INL) report, *Component Age Traits from EPIX*, July 3, 2012. The INL report also evaluated the extent to which age-related active component failures were increasing, but Office of Nuclear Regulatory Research staff could not determine from the INL report if that was the case, due to limitations with the available data.

In another active component age-related project, regional office inspection staff familiar with active component aging problems and the *IOEB Study* conducted what staff described as an inspection to determine if active component aging could be addressed through existing inspection procedures. Accordingly, a Problem Identification and Resolution inspection was scheduled and conducted in November 2012 with the objective of gathering information to determine if a licensee had a periodic, time-based replacement program for aging active components.

[4] This study is publically available; see ADAMS accession number ML13044A469.
[5] These data were originally sourced from the Institute for Nuclear Power Operations Equipment Performance and Information Exchange database.

Regulations Used for Citing Licensees for Age-Related Degradation

Inspectors can use various NRC regulations to cite licensees for age-related degradation of active components that are not specific to aging. These regulations do not establish limits on the age of active components in commercial nuclear power plants, or prohibit degradation of active components by aging. Instead, NRC has regulations that establish equipment performance requirements that may not be met by components that have degraded due to aging. Inspectors said that they use the following regulations for inspections to meet the challenge of identifying aging active components and citing licensees for age-related violations:

- 10 CFR 50.65, *Requirements for monitoring the effectiveness of maintenance at nuclear power plants* (the Maintenance Rule).
- 10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*.
- 10 CFR 50.36, *Technical specifications*.

*The Maintenance Rule*

NRC inspectors' experience with citing licensees for violations of the Maintenance Rule for age-related active component failures varies because—as NRR and regional inspection staff explained—the Maintenance Rule is not well understood by less experienced NRC staff. Furthermore, NRR and regional staff said that there are not enough NRC staff with Maintenance Rule experience who can use it to cite licensees for age-related failures within the performance-based ROP framework. NRC management and staff also conveyed that an experienced inspector may cite a licensee for age-related inspection findings and violations under the Maintenance Rule Section (a)(3) when the licensee has not taken into consideration applicable operating experience, whereas less experienced inspectors are less likely to do so. Other NRC staff voiced a similar concern regarding a lack of inspectors with experience using the Maintenance Rule. OIG noted that inspectors have rarely used the Maintenance Rule to cite licensees for failure of safety-related components due to aging. Of the 105 age-related active component failures and reportable events from 2007 to 2011 identified in the *IOEB Study*, only 3 were cited for violations of the Maintenance Rule. Although Maintenance Rule violations do include some findings attributed to aging,

most Maintenance Rule violations do not include end of life aging as defined by the *IOEB Study*.

The following table provides additional details relating to the three Maintenance Rule violations noted above. In each case, the licensee left active components in service until they failed through shortcomings in preventive maintenance activities.

**Figure 1: Three NRC Maintenance Rule Citations, 2007-2011**

| Date | Plant | Type of Age-Related Failure | Cause |
|---|---|---|---|
| 2011 | Waterford | Electronic control components for cooling towers failed after operating for 25 years. | Aging. The maintenance to replace the components was deleted from preventive maintenance activities. |
| 2009 | Catawba | Auxiliary feed water sump valves important to plant safety failed. | Aging. No maintenance was performed on the valves since plant startup in 1985. |
| 2007 | Brunswick | A relay for controlling an emergency diesel generator failed. | Aging. The relay's coil failed due to the deferral of maintenance. |

Source: *IOEB Study*

According to agency staff, the ROP's emphasis on performance-based and risk-informed oversight limits their ability to evaluate licensee preventive maintenance programs. This limitation exists because NRC does not perform programmatic inspections of licensees' preventive maintenance programs. For example, a regional senior management

official described how there were aging problems found at Fort Calhoun by the inspectors during additional inspections after a flood, but routine ROP inspections are not structured to focus inspection resources on active component age-related issues. Another regional senior management official confirmed that the ROP does not have provisions to look at a licensee's preventive maintenance programs, but focuses on performance and evaluates licensee response to failures after the fact—that is, after a component failure—and how the licensee addresses the cause of the failure as part of its Corrective Action Program.

Several NRC staff with inspection experience stated that the Maintenance Rule as currently in use is not structured to address aging active components.

*Other Regulations Used for Citing Licensees for Age-Related Failures*

Inspectors are likely to use other regulatory provisions—specifically, 10 CFR Part 50, Appendix B, and plant technical specifications—for citing licensees for age-related failures because findings can be supported and justified more readily than by using the Maintenance Rule. According to NRC staff, using 10 CFR Part 50, Appendix B, provisions to cite licensees for quality assurance shortcomings related to component design, fabrication, construction, operation, and testing is easier than using the Maintenance Rule. Similarly, NRC inspectors have cited licensees for violating plant technical specifications, specifically for lacking adequate justification for running active components beyond vendor recommended life and not having appropriate maintenance procedures in place. However, 10 CFR Part 50, Appendix B, and 10 CFR 50.36, *Technical specifications*, do not contain age-related criteria for supporting age-related findings.

Management Not Always Cognizant of Active Aging Component Activities

Agency managers are not always cognizant of the staff's activities related to active component aging. NRR senior managers were not aware of the status of staff implementation of the *IOEB Study* conclusions and recommendations. Although senior managers indicated that NRC has robust programs in place that address active component aging and that staff were following up on the *IOEB Study* conclusions and recommendations, neither was the case during the course of the audit.

Staff indicated that the agency is taking action to follow up on the results of the aging component study, but the agency's plan for considering and implementing the *IOEB Study* conclusions and recommendations is unclear.

Although the *IOEB Study* was peer-reviewed and issued internally via OpE COMM[6] in June 2012 and presented to an ROP enhancement group in March 2013, a number of senior managers and program office and regional staff having active component aging responsibilities were unfamiliar with the status of staff implementation of the *IOEB Study* conclusions and recommendations. One NRR senior manager indicated that the *IOEB Study* was an important effort and thought that agency staff were following up on the study's conclusions and recommendations. However, the NRR senior manager's direct report and regional senior managers that the NRR senior manager identified as familiar with the *IOEB Study* were not aware of the details of the study or proposed followup actions.

NRR senior management also indicated that regional office management officials would provide insight on aging active component efforts and follow up on the *IOEB Study* conclusions and recommendations. However, of the two regional management officials to whom NRC staff had provided copies of the report, one noted that the *IOEB Study* was for informational purposes only and did not require action. Furthermore, neither was aware of the details of the *IOEB Study* and therefore was not able to offer insight on aging active component efforts and staff followup on the *IOEB Study* conclusions and recommendations.

Furthermore, agency managers and staff have conflicting understandings of and did not coordinate how a biennial ROP enhancement effort[7] may incorporate changes to active component aging oversight. Specifically, senior managers indicated that *IOEB Study* conclusions and

[6] OpE COMMs (Operating Experience Communications) contain preliminary information in the interest of timely internal communication of operating experience. OpE COMMs may be predecisional and may contain sensitive information. They are not intended for distribution outside the agency.

[7] The biennial ROP review allows NRC to evaluate licensee performance on a regular, recurring basis. In 2013, NRC undertook to enhance the review by taking additional measures as part of the review. During the review, named ROP Enhancement—Baseline Inspection Program, champions and key staff will make changes to the inspection procedures based on analysis completed by the inspection procedure owners, information and knowledge from inspectors, special groups and reports, lessons learned, and recent events and inspections.

recommendations will likely be included in the 2013 ROP enhancement effort. However, a key NRR staff member involved with the review said that ROP enhancement will not include active component aging because the Maintenance Rule, in the staff's opinion, already addresses active component aging. Yet staff in the IOEB stated that, as far as they were aware, there had been no official coordination between the operating experience and the inspection branches for including active component aging oversight. Nonetheless, NRC staff were unable to provide any specific documentation of the agency's plan for including the *IOEB Study* conclusions and recommendations in the ROP enhancement effort. Consequently, OIG could not independently confirm the extent to which the subject of active component aging is being considered for inclusion in the review.

## NRC Has Not Evaluated or Analyzed the Need for a Formal Program and Has Not Systematically and Continually Collected or Evaluated Active Component Age-Related Data

The unfocused and uncoordinated approach NRC uses in its oversight of licensees' active aging component activities is occurring because NRC:

- Has not conducted a systematic evaluation of program needs for overseeing licensees' aging management for active components.

- Does not have mechanisms for systematic and continual monitoring, collecting, and trending of age-related data for active components.

### NRC Has Not Conducted a Systematic Program Needs Evaluation and Analysis

Since the ROP was initiated in 2000, NRC has not conducted an evaluation and analysis that would systematically determine whether the need exists for a formal active aging component oversight program. NRC has not systematically evaluated the need for specific program policies, goals, and objectives, and the need for program feedback and corrective actions for continual improvement, all within the context of the current ROP environment.[8] The most recent evaluation of the agency's regulatory

[8]This is not to suggest that NRC inspection procedures have remained static over the years. For example, according to agency staff, Inspection Procedure 71111.21, *Component Design Bases Inspection*, was updated in August 2012

oversight of active component aging was in 1996[9]—which pre-dates the ROP—and stressed the importance of aging studies as an important part of efforts to identify and solve potential aging problems.

In addition, nuclear plants have aged almost 20 years since the most recent evaluation of the agency's regulatory oversight of active component aging in 1996. The report that resulted from this evaluation asserted that,

> …active components generally do not present a significant aging problem in nuclear power plants. Design criteria and effective preventative maintenance programs, including timely replacement of components, are effective in mitigating potential aging problems… .

However, NRC does not inspect preventive maintenance programs directly and comprehensively to ensure they are effective. Aging can cause active component degradation and failure resulting in unexpected reactor power changes, failures of components to perform their safety function, and adverse effects to the safety margin. For example:

- OIG identified an unexpected reactor power change and automatic reactor shutdown that occurred at a commercial power reactor in 2012. The shutdown was due to a failed switch which had been in service over 40 years with no preventive maintenance performed.

- The *IOEB Study* reported an incident from 2010 whereby a relay failed after not being replaced or monitored at the required periodicity, resulting in the failure of an emergency diesel generator to provide power to safety systems when it was called upon to do so during a partial loss of electrical power at the plant.

to include the statement "…inspectors should try to determine through review of these corrective work maintenance activities whether licensee's preventive maintenance or other programs such as aging management are being reasonably effective in preventing component failures." OIG's wider point is that activities—such as inspection procedure changes—have been undertaken without the benefit of a systematic program needs evaluation and analysis.

[9] NUREG/CR-6442, *Evidence of Aging Effects on Certain Safety-Related Components,* NRC and Idaho National Laboratories, 1996.

- The *IOEB Study* also reported that in 2010, an age-related failure of a pressure switch caused an unexpected reactor power change and automatic reactor shutdown. This pressure switch had been installed for 39 years.

Other active component aging studies conducted both in the United States and internationally offer examples of qualitative and quantitative data related to active component failures. In general, these studies all emphasize the importance of having a continual awareness of potential aging problems.

NRC Has Not Systematically and Continually Collected or Evaluated Active Component Age-Related Data

NRC has not developed and incorporated within policy and guidance the existing mechanisms used for systematic and continual monitoring, collecting, and trending of age-related data for active components. Age-related studies have emphasized the importance of continual monitoring, collecting, and trending of age-related data for active components in an ever changing environment. Yet, NRC has not systematically and continually collected or evaluated age-related data to determine if a specific oversight program is needed or what type of program would be necessary. Currently, NRC may identify data on active component aging intermittently during ROP inspections, but not through any methods of systematic data collection, analysis, and trending. Although the *IOEB Study* and the INL report identified age-related data from existing reports and evaluated it based on types of failure and age-related causes, discussions with NRC staff revealed that at present, age-related failures are not consistently identified in existing reporting mechanisms, when they are identified at all. An experienced NRR staff member told OIG that there could be a rise in the number of identified age-related events if the proper regulatory tools were in place to identify them.

## NRC Cannot Be Fully Assured of Effective Oversight

Despite management's belief that active component aging issues are being satisfactorily addressed, NRC is not in a position to draw any conclusions one way or the other. If NRC's unfocused and uncoordinated approach for oversight of licensees' active component aging activities

continues, NRC will not be fully assured that it is effectively overseeing licensees' aging active component programs. Specifically, the agency will not be a position to:

- Identify and evaluate trends that have safety implications.
- Proactively identify active components subject to age-related failure before they are run to failure.
- Provide complete inspector training and guidance.
- Close the performance gap between experienced inspectors who know how to identify active components that are run beyond their reasonably expected service lives and less experienced inspectors who must evaluate active component failures where age degradation may have been a significant factor.

**Conclusions**

NRC's unfocused and uncoordinated approach for oversight of active aging component activities is characterized by staff-initiated projects and inspection activities, and incognizant managers. Without direction from senior management, staff are conducting work to heighten awareness of active component aging and senior management is not aware of various active component aging oversight activities that are underway.

Despite concerns of component aging in nuclear power plants that are growing older, the agency does not routinely collect and monitor instances of active component failures due to aging. Indeed, the very act of inspecting for these aging effects *before* failures occur appears to be difficult for the agency to undertake under ROP. This failure to routinely collect and monitor such data runs counter to the need to do so for the agency to maintain and adjust its approach to active component aging oversight.

## Recommendations

OIG recommends that the Executive Director for Operations:

1. Perform and document a thorough and systematic evaluation of the need for an NRC program to oversee the management of active component aging activities, all within the context of the current ROP environment. Evaluation elements are to include, but should not be limited to, the need for:

   (a) Program policies, goals, and objectives.

   (b) Program feedback and corrective actions for continual improvement.

2. Develop and incorporate the mechanisms for monitoring, collecting, and trending age-related data for active components within NRC policy and procedures.

# IV. AGENCY COMMENTS

On July 22, 2013, OIG issued the discussion draft of this report to the Executive Director for Operations. OIG met with NRC management and staff on August 20, 2013, at an exit conference and on August 26, 2013 at a staff meeting to discuss the draft report content. At these meetings, the agency provided informal comments, which OIG subsequently incorporated into the draft report as appropriate.

On September 27, 2013, NRC provided formal comments to the draft report that indicated their disagreement with the audit report content. The agency's formal comments state, in part, that NRC disagrees that it needs to perform a thorough and systematic evaluation to determine the need for a specific NRC program to oversee the management of active component aging activities, because the ROP performs this task by providing a framework for ensuring that both active and passive aging issues are addressed. OIG auditors concluded that the agency is not in a position to determine the effectiveness of active component aging oversight. This is because the agency uses regulations and inspection procedures for oversight of active component aging that were established prior to the implementation of ROP in 2000 and has not, since ROP implementation, evaluated whether those regulations and inspection procedures work the same way as intended in the pre-ROP regulatory environment.

Appendix A contains the audit Objective, Scope and Methodology; Appendix B contains a copy of the agency's formal comments; and Appendix C contains OIG's analysis of the agency's formal comments.

APPENDIX A

# OBJECTIVE, SCOPE, AND METHODOLOGY

## OBJECTIVE

The audit objective was to determine if NRC is providing effective oversight of industry's aging component programs.

## SCOPE

We conducted this performance audit at NRC headquarters in Rockville, MD, and collected information from the regional offices via telephone and in conjunction with the Audit of NRC's Support for Resident Inspectors, from October 2012 through May 2013. The audit scope was limited to NRC's regulatory responsibilities as they pertain to aging active component programs at commercial nuclear power plants. Internal controls related to the audit objectives were reviewed and analyzed.

## METHODOLOGY

To address the audit objective, OIG interviewed agency senior management officials, and headquarters and regional staff. OIG also reviewed NRC regulations and guidance, as well as domestic and international operational experience reports pertaining to active aging components. OIG subsequently compared the information provided during the interviews with staff actions. Throughout the audit, auditors were aware of the possibility or existence of fraud, waste, or misuse in the program. Some of the key documents referred to in this report include the following:

Regulations:

- 10 CFR Part 50.65, *Requirements for monitoring the effectiveness of maintenance at nuclear power plants*.
- 10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*.
- 10 CFR Part 50.36, *Technical specifications*.

Inspection Procedures:

- Inspection Procedure 71111.21, *Component Design Bases Inspection*.
- Inspection Procedure 71152, *Problem Identification and Resolution*.

Operating Experience Reports:

- *IOEB Component Aging Study (2007-2011) – Insights from Inspection Findings and Reportable Events,* July 24, 2012.
- *IOEB Analysis Team Study on Recent Operating Experience Ineffective Use of Vendor Technical Recommendations*, June 6, 2011.
- Idaho National Laboratories, *Component Age Traits from EPIX*, July 3, 2012.
- International Atomic Energy Agency Safety Report Series, No. 62, *Proactive Management of Ageing for Nuclear Power Plants*, 2009.
- European Commission Joint Research Center, *Ageing Related Events Topical Study*, 2011.
- NRC and Idaho National Laboratories, *Evidence of Aging Effects on Certain Safety-Related Components,* NUREG/CR-6442, 1996.

We conducted this performance audit in accordance with generally accepted government auditing standards. Those standards require that we plan and perform the audit to obtain sufficient, appropriate evidence to provide a reasonable basis for our findings and conclusions based on our audit objectives. We believe that the evidence obtained provides a reasonable basis for our findings and conclusions based on our audit objectives.

Contributors to this report were R.K. Wild, Team Leader; Kevin Nietmann, Senior Technical Advisor; Vicki Foster, Audit Manager; Timothy Wilson, Senior Management Analyst; Larry Weglicki, Senior Auditor; Jenny Cheung, Auditor; and Tariq Noaman, Management Analyst.

**APPENDIX B**

# AGENCY FORMAL COMMENTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 27, 2013

MEMORANDUM TO: Stephen D. Dingbaum
Assistant Inspector General for Audits
Office of the Inspector General

FROM: Michael R. Johnson
Deputy Executive Director for Reactor
and Preparedness Programs
Office of the Executive Director of Operations

SUBJECT: FORMAL COMMENTS ON OFFICE OF THE INSPECTOR GENERAL DRAFT REPORT ON THE NUCLEAR REGULATORY COMMISSION'S OVERSIGHT OF ACTIVE COMPONENT AGING

This memorandum responds to Ms. Vicki Foster's September 13, 2013, e-mail transmitting the Office of the Inspector General's (OIG) revised draft report, "Audit of the U.S. Nuclear Regulatory Commission's Oversight of Active Component Aging." Based on a thorough review of the draft audit report and extensive interaction between the OIG and U.S. Nuclear Regulatory Commission (NRC) staff, the NRC disagrees with the major premise of this audit, which is that the NRC oversight of active component aging is not being effectively dealt with under existing oversight programs. Nearly all of the active component aging data contained within the Office of Nuclear Reactor Regulation (NRR) Operating Experience Branch (IOEB) Aging Study was derived from inspection findings in which NRC inspectors effectively documented active component aging issues and cited associated violations of the regulations.

The staff also does not agree that its activities for managing active component aging are not focused or coordinated. While we agree that there is no one section of the regulations or oversight programs that specifically deals with active component aging, and while there is always room for improvement, active component aging issues are effectively addressed through various aspects of the regulations and oversight programs. For example, the staff relies on Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (Criteria III, V, XI, and XVI), technical specifications, and the Maintenance Rule, all of which deal with component degradation and age-related performance issues.

Licensee performance is assessed against these regulations under the Reactor Oversight Process (ROP), which allows inspectors and NRC managers to ensure regulatory compliance. The NRC deals with issues identified under the ROP in a performance-based and risk-informed manner. Furthermore, the NRC has ample regulations that require licensees to report performance issues that may be caused by active component aging. These include the

CONTACT: John W. Thompson, NRR/DIRS
301-415-1011

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reporting requirements of 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Plants," and 10 CFR 50.73, "Licensee Event Report System."

When this information is combined with ROP inspection results and industry component failure databases—all of which are routinely assessed by the staff—they provide ample visibility for the staff to collect and evaluate age-related data for active components. The staff does not believe that additional programs that involve monitoring, collecting, and trending of active component aging are necessary to identify adverse trends or to take appropriate regulatory action.

The staff also disagrees that it needs to perform a thorough and systematic evaluation to determine the need for a specific NRC program to oversee the management of active component aging activities. The ROP performs this task by providing a framework for ensuring that both active and passive aging issues with the potential to impact safety are addressed in a timely manner. The NRC has processes in place to systematically evaluate the results of the ROP, along with other data sources, and these processes provide adequate assurance that the NRC will identify safety-significant, age-related failures of active components.

Furthermore, the staff believes that no additional mechanisms are needed to monitor, collect, and trend age-related data on active component aging. The staff already collects or has access to operating experience data gathered from reportable events, international events, industry failure data, and inspection findings that are routinely screened for significance and trending and analysis. The creation of additional mechanisms to perform these tasks is not necessary.

Other staff comments on the OIG audit are as follows:

- There are valid reasons why inspectors cite different requirements to document findings involving age-related issues. These may include variability in the identified performance deficiencies, event causes, and plant licensing bases. Reasons for the variability in citing different requirements were not explored in the report.

- The staff is actively considering recommendations identified in the IOEB study for improving inspector awareness of aging issues and clarifying inspection procedure guidance as part of the ongoing ROP Enhancement Project.

- The IOEB study represented a new approach for performing analysis of ROP data. While the program has experience with the communication and implementation of recommendations for specific technical issues, the structure for implementing the kind of broad recommendations that were presented in this study, and which require coordination across multiple offices, is being refined as the recommendations are being implemented, presenting learning opportunities and challenges for their completion.

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- The NRC staff points to the overall capability of the ROP (including the structure of the Significance Determination Process) and current reporting requirements as substantive evidence that there is adequate assurance that it is addressing active component aging issues impacting safety.

cc: Chairman Macfarlane
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
SECY

**APPENDIX C**

# OIG ANALYSIS OF AGENCY FORMAL COMMENTS

On July 22, 2013, OIG issued the discussion draft of this report to the Executive Director for Operations. OIG met with NRC management and staff on August 20, 2013, at an exit conference and on August 26, 2013, at a staff meeting to discuss the draft report content. At these meetings, the agency provided informal comments, which OIG subsequently incorporated into the draft report as appropriate. On September 27, 2013, NRC provided formal comments to the draft report that indicated their disagreement with the audit report contents. OIG's analysis of those comments is as follows:

OIG maintains that NRC needs to improve the unfocused and uncoordinated approach management officials use for oversight of licensees' management of active component aging. The comments that the agency provided reflect an overall misunderstanding and misinterpretation of the report content.

OIG has assigned a reference number to each of the agency's comments to aid the following point-by-point analysis:

Agency Comment 1:

"NRC disagrees with the major premise of this audit, which is that the NRC oversight of active component aging is not being effectively dealt with under existing oversight programs."

OIG Response to Agency Comment 1:

The agency misunderstands and has misinterpreted the audit's major premise. The major premise of this report is not that oversight of active component aging is not being effectively dealt with under existing oversight programs. The message is that OIG could not determine the extent to which NRC provides effective oversight due to a lack of agency analysis that tests its assumptions regarding active component aging. OIG concluded that this has occurred because NRC has not conducted a systematic evaluation of program needs for overseeing licensees' aging management for active components since the establishment of ROP in 2000, and does not have mechanisms for continual monitoring and trending of age-related data for active components. Consequently, NRC cannot be fully assured that it is effectively overseeing licensees' management of aging active components.

Agency Comment 2:

"Nearly all of the active component aging data contained within the Office of Nuclear Reactor Regulation (NRR) Operating Experience Branch (IOEB) Aging Study was derived from inspection findings in which NRC inspectors effectively documented active component aging issues and cited associated violations of the regulations."

OIG Response to Agency Comment 2:

NRC's statement that nearly all of the active component aging data contained within the Office of Nuclear Reactor Regulation *IOEB Component Aging Study 2007-2011 — Insights from Inspection Findings and Reportable Events*, July 24, 2012 (*IOEB Study*) was derived from inspection findings in which NRC inspectors effectively documented active component aging issues and cited associated violations of the regulations is not correct.

OIG analysis of the inspection and event reports from which the *IOEB Study* data was derived indicates that about one quarter of the failures identified in the report were not derived from inspection reports. Rather, these failures were provided by licensees to NRC in licensee event reports. Moreover, almost all of the regulatory non-compliances are non-cited violations. The agency's claim that almost all of the 105 events were cited violations derived from inspection activity is inaccurate. In fact, more than half of the events in the report did not result in violations of any kind.

Prior OIG analysis[10] indicates that for every safety-related component failure that occurs and is reported, there are several that occur but do not meet the level of reportability using 10 CFR Part 50.72, *Immediate Notification Requirements for Operating Nuclear Power Plants* and 10 CFR 50.73, *Licensee Event Report System* reporting criteria, indicating that there may be many more safety-related components that have degraded or failed due to aging that have not been reported to NRC.

[10] *Audit of NRC's Implementation of 10 CFR Part 21, Reporting of Defects and Noncompliance*, OIG-11-A-08, March 23, 2011.

Agency Comment 3:

"The staff also does not agree that its activities for managing active component aging are not focused or coordinated. While we agree that there is no one section of the regulations or oversight programs that specifically deals with active component aging, and while there is always room for improvement, active component aging issues are effectively addressed through various aspects of the regulations and oversight programs. For example, the staff relies on Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (Criteria III, V, XI, and XVI), technical specifications, and the Maintenance Rule, all of which deal with component degradation and age related performance issues."

OIG Response to Agency Comment 3:

OIG did not conclude that regulations were unfocused and uncoordinated or that active component aging issues are not effectively addressed through various aspects of the regulations and oversight programs. Rather, OIG concluded that NRC's approach for oversight of licensees' management of active component aging is not focused or coordinated because NRC's approach includes staff-initiated projects, using inspections that are not aging-related, and agency senior managers who are not aware of these uncoordinated activities.

Furthermore, NRC asserted that active component aging issues are effectively addressed through various aspects of the regulations and oversight programs. For the sake of clarity, OIG did not indicate anywhere in the report that the use of the regulations was ineffective. OIG even stated in the report that inspectors have used those regulations to support a basis for age-related inspection findings. However, with regard to effectiveness, OIG stated in the report that it could not determine the extent to which NRC provides effective oversight due to a lack of agency analysis that tests its assumptions regarding active component aging.

Agency Comment 4:

"Licensee performance is assessed against these regulations under the Reactor Oversight Process (ROP), which allows inspectors and NRC managers to ensure regulatory compliance. The NRC deals with issues identified under the ROP in a performance based and risk informed manner. Furthermore, the NRC has ample regulations that require licensees to report performance issues that may be caused by active component aging. These include the reporting requirements of 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Plants," and 10 CFR 50.73, "Licensee Event Report System."

OIG Response to Agency Comment 4:

The agency states that NRC has ample regulations that require licensees to report performance issues that may be caused by active component aging. However, OIG did not conclude that regulations or reporting methods were inadequate. OIG did note that NRC was not in a position to determine whether or not oversight of active aging components was effective because NRC had not collected and evaluated the requisite data to determine the need for further action regarding active component aging. In such a scenario, it would be inappropriate for OIG, NRC, or anybody else to conclude one way or the other whether the regulations or reporting methods were adequate.

Agency Comment 5:

"When this information is combined with ROP inspection results and industry component failure databases—all of which are routinely assessed by the staff—they provide ample visibility for the staff to collect and evaluate age related data for active components. The staff does not believe that additional programs that involve monitoring, collecting, and trending of active component aging are necessary to identify adverse trends or to take appropriate regulatory action."

OIG Response to Agency Comment 5:

NRC states that the staff does not believe that additional programs that involve monitoring, collecting, and trending of active component aging are necessary to identify adverse trends or to take appropriate regulatory action. OIG recognizes that NRC collects a great deal of industry operating experience. However, the agency does not collect or evaluate it for aging active component degradation or failures. Given the evidence reviewed in this audit, OIG concluded that NRC should establish a factual basis for its belief that no additional programs for the monitoring, collecting, and trending of active component aging data are necessary.

Agency Comment 6:

"The staff also disagrees that it needs to perform a thorough and systematic evaluation to determine the need for a specific NRC program to oversee the management of active component aging activities. The ROP performs this task by providing a framework for ensuring that both active and passive aging issues with the potential to impact safety are addressed in a timely manner. The NRC has processes in place to systematically evaluate the results of the ROP, along with other data sources, and these processes provide adequate assurance that the NRC will identify safety-significant, age-related failures of active components."

OIG Response to Agency Comment 6:

As noted in the report, the agency is not in a position to determine the effectiveness of active aging oversight because they have not done the necessary evaluation to conclude whether or not an active component aging oversight program is needed. OIG asked for, and the agency did not provide, evidence that it already performs a systematic evaluation that proves specifically how ROP and the other processes adequately address active component aging.

Furthermore, for ROP and the other processes that NRC asserts adequately address active component aging, the agency does not evaluate program results to determine if active component aging degradation and failures are acceptable or not. A recent example—whereby inspectors found numerous active component age-related issues during additional inspections in response to failures at the Fort Calhoun Nuclear Power Plant after notable performance deficiencies occurred—suggests the need to evaluate all of those regulations and processes that NRC claims have been effective in the oversight of active component aging.

Agency Comment 7:

"Furthermore, the staff believes that no additional mechanisms are needed to monitor, collect, and trend age-related data on active component aging. The staff already collects or has access to operating experience data gathered from reportable events, international events, industry failure data, and inspection findings that are routinely screened for significance and trending and analysis. The creation of additional mechanisms to perform these tasks is not necessary."

OIG Response to Agency Comment 7:

The staff believes that no additional mechanisms are needed to monitor, collect, and trend age-related data on active component aging because the staff already collects or has access to operating experience data. In fact, OIG found that the staff does collect data, but the data is not analyzed to identify or trend active component degradation/failures. Furthermore, OIG notes that existing operating experience data gathered from reportable events, international events, industry failure data, and inspection findings are routinely screened for significance and trending and analysis but not for active component degradation and failure due to aging. NRC does not collect active aging data using "active component aging" as a specific, discrete subcategory, although the infrastructure is in place to do so. According to an industry organization representative's presentation during the 2013 NRC Regulatory Information Conference, the system NRC uses to obtain age-related data could be programmed to include aging as a cause code field that can be selected for age-related data collection purposes. Furthermore,

an NRR senior manager said that he would communicate with an industry contact to obtain NRC access to additional operating experience data.

OIG determined that NRC has not developed or incorporated within policy and guidance the mechanisms for continually monitoring, collecting, and trending age-related data for active components. These mechanisms could be the existing infrastructure for collecting and evaluating operating experience data or new mechanisms at NRC's discretion.

OIG made changes to this report to clarify that NRC has not systematically and continually collected or evaluated active component age-related data using existing mechanisms.

Agency Comment 8:

"There are valid reasons why inspectors cite different requirements to document findings involving age-related issues. These may include variability in the identified performance deficiencies, event causes, and plant licensing bases. Reasons for the variability in citing different requirements were not explored in the report."

OIG Response to Agency Comment 8:

OIG *described* how inspectors cite licensees against regulations to address aging active component failures and did not challenge the use of multiple regulations to cite licensees. OIG did not state that the reasons why inspectors cite different requirements to document findings involving age–related issues was invalid or question the variability in citing different requirements. OIG *does* call into question, however, the variability in use of any of the regulatory or inspection tools that is due to unfamiliarity or lack of inspector experience as described in this report with regard to a lack of understanding of the Maintenance Rule for purposes of inspecting for active component aging phenomena. That such variability exists for these reasons and that NRC managers are unaware of this variability is a central message of the OIG report.

Agency Comment 9:

"The staff is actively considering recommendations identified in the *IOEB Study* for improving inspector awareness of aging issues and clarifying inspection procedure guidance as part of the ongoing ROP Enhancement Project."

OIG Response to Agency Comment 9:

The *IOEB Study* was completed in July 2012 and was presented to NRR management in November 2012. In March 2013, NRC staff provided OIG with

conflicting information regarding inclusion of the aging study as part of the ROP Enhancement Project, which was documented in the audit report. OIG continues to note that NRC is actively considering the *IOEB Study* recommendations; however, the agency has not been clear as to what, exactly, comprises that consideration.

Agency Comment 10:

"The IOEB study represented a new approach for performing analysis of ROP data. While the program has experience with the communication and implementation of recommendations for specific technical issues, the structure for implementing the kind of broad recommendations that were presented in this study, and which require coordination across multiple offices, is being refined as the recommendations are being implemented, presenting learning opportunities and challenges for their completion."

OIG Response to Agency Comment 10:

The agency states that it is in the process of implementing the *IOEB Study* recommendations. However, OIG notes that this comment is inconsistent with Agency Comment 9 above that states the *IOEB Study* is being considered as part of the ROP Enhancement Project. Based on these formal comments, it is not clear whether the agency is considering or implementing the recommendations. The agency has been rather opaque in its responses to OIG auditor inquiries as to what recommendations, specifically, are under consideration or being implemented.

Agency Comment 11:

"The NRC staff points to the overall capability of the ROP (including the structure of the Significance Determination Process) and current reporting requirements as substantive evidence that there is adequate assurance that it is addressing active component aging issues impacting safety."

OIG Response to Agency Comment 11:

The agency did not provide substantive evidence of adequate assurance that it is addressing active component aging issues impacting safety. In fact, the agency provided no evidence and believes that active component issues are addressed through ROP. OIG could not therefore determine the extent to which NRC provides effective oversight due to a lack of agency analysis that tests its assumptions regarding active component aging.

Furthermore, OIG did not state that there was inadequate assurance that ROP and current reporting requirements were addressing active component aging

issues impacting safety. OIG noted that a systematic evaluation of program needs for overseeing licensees' aging management for active components has not been conducted since the establishment of ROP in 2000. Additionally, NRC does not trend active component aging degradation and failures that could provide evidence of adequate assurance that it is addressing active component aging issues impacting safety. Consequently, NRC cannot be fully assured that it is effectively overseeing licensees' management of aging active components.

# EXHIBIT 22

# AUDIT REPORT

Audit of NRC's Management of Licensee Commitments

OIG-A-17 September 19, 2011

All publicly available OIG reports are accessible through
NRC's Web site at:
http:/www.nrc.gov/reading-rm/doc-collections/insp-gen/

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OFFICE OF THE
INSPECTOR GENERAL

September 19, 2011

MEMORANDUM TO: R. William Borchardt
Executive Director for Operations

FROM: Stephen D. Dingbaum **/RA/**
Assistant Inspector General for Audits

SUBJECT: AUDIT OF NRC'S MANAGEMENT OF LICENSEE COMMITMENTS (OIG-11-A-17)

Attached is the Office of the Inspector General's (OIG) audit report titled, *Audit of NRC's Management of Licensee Commitments.*

The report presents the results of the subject audit. Agency comments provided at a meeting with NRC management officials and staff on August 12, 2011, and an August 23, 2011, exit conference have been incorporated, as appropriate, into this report.

Please provide information on actions taken or planned on each of the recommendations within 30 days of the date of this memorandum. Actions taken or planned are subject to OIG follow up as stated in Management Directive 6.1.

We appreciate the cooperation extended to us by members of your staff during the audit. If you have any questions or comments about our report, please contact me at 415-5915 or RK Wild, Team Leader, Nuclear Reactor Safety Team, at 415-5948.

Attachment: As stated

Electronic Distribution

Edwin M. Hackett, Executive Director, Advisory Committee
on Reactor Safeguards
E. Roy Hawkens, Chief Administrative Judge, Atomic Safety
and Licensing Board Panel
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Elmo E. Collins, Jr., Regional Administrator, Region IV

# EXECUTIVE SUMMARY

## BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC) regulates commercial nuclear power plants that generate electricity through a combination of regulatory requirements and licensing, inspection, and enforcement activities. One way NRC provides oversight of licensees is through the management of regulatory commitments (commitments).

Commitments are docketed, written statements describing a specific action that the licensee has agreed or volunteered to take. They often result from a licensing action such as a license amendment, including power uprates, or from a generic communication, such as generic letters and bulletins. Commitments are neither legally binding nor obligations of a license; however, a commitment may be escalated into a legally binding obligation only if NRC staff deems that the commitment is essential for ensuring public health and safety.

Licensees are responsible for creating, tracking, and handling all commitments made to NRC. The licensee is entirely responsible for tracking the commitments, and this includes any changes to the commitments and notification to NRC about such changes. NRC expects licensees to honor commitments in good faith.

## PURPOSE

The audit objective was to assess the extent to which NRC appropriately and consistently utilizes and manages regulatory commitments for power reactor licensees.

## RESULTS IN BRIEF

Part of NRC's mission is to identify and accomplish those actions that provide the level of nuclear plant performance necessary to ensure adequate protection of public health and safety. A commitment is one tool that NRC uses in the overall licensing process to add flexibility, improve efficiency, and maintain the flow of information between the staff and

licensees. OIG identified opportunities for improvement in the following three areas:

- Consistent implementation of commitment management audits.
    - NRC inconsistently implements the audits of licensee commitment management programs. This is because agency guidance concerning its performance of required triennial audits is incomplete and imprecise. Incomplete and imprecise guidance concerning the conduct of commitment management audits can result in ineffective audits, inefficient use of resources, and the appearance that NRC provides disparate oversight of similarly situated licensees.
- Staff understanding of the definition and use of commitments.
    - The definition and use of commitments is not consistently understood throughout the agency. This occurs because NRC training on commitments is insufficient. Specifically, training does not effectively address the definition and use of commitments and is not provided to all agency staff involved in reviewing licensee commitments. This could potentially result in the misapplication of commitments by NRC staff.
- NRC tracking of commitments.
    - NRC does not systematically track commitments because the agency does not have an adequate tool for tracking them, in part, because the agency has not identified a need for such a tool. Consequently, NRC cannot completely ensure oversight of commitments, which has implications for the agency's continuing awareness of significant commitments, the effectiveness of the triennial commitment management audits, and institutional knowledge management.

## RECOMMENDATIONS

This report makes five recommendations to improve the agency's management of licensee commitments. A consolidated list of these recommendations appears in Section V of this report.

## AGENCY COMMENTS

On August 9, 2011, the Office of the Inspector General (OIG) issued the discussion draft of this report to the Executive Director for Operations. OIG met with NRC management officials and staff on August 12, 2011, at which time the agency provided informal comments to the draft report. OIG subsequently met with agency management and staff during an August 23, 2011, exit conference to discuss agency informal comments that OIG incorporated into the draft report as appropriate. At this meeting, agency management provided supplemental information that has also been incorporated into this report as appropriate. NRC management and staff reviewed the revised draft report and generally agreed with the recommendations in this report. Furthermore, the agency opted not to provide formal comments for inclusion in this report.

## ABBREVIATIONS AND ACRONYMS

| | |
|---|---|
| ADAMS | Agencywide Documents Access and Management System |
| CFR | Code of Federal Regulations |
| DORL | Division of Operating Reactor Licensing |
| FSAR | Final Safety Analysis Report |
| MD | Management Directive |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| NRR | Office of Nuclear Reactor Regulation |
| OGC | Office of the General Counsel |
| OIG | Office of the Inspector General |
| UFSAR | Updated Final Safety Analysis Report |

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# I. BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC) regulates commercial nuclear power plants that generate electricity through a combination of regulatory requirements and licensing, inspection, and enforcement activities. One way NRC provides oversight of licensees is through the management of regulatory commitments (commitments), which are non-legally binding actions that the licensee agrees or volunteers to take. Licensees are responsible for creating, tracking, and handling all commitments made to NRC. Within NRC, the primary responsibility for managing commitments lies with Division of Operating Reactor Licensing (DORL) project managers in the Office of Nuclear Reactor Regulation (NRR). However, other NRC staff—such as DORL branch chiefs, NRR technical reviewers, and Office of the General Counsel (OGC) staff—are involved in decisionmaking processes that use commitments.

**Guidance on Commitments**

Although the term "regulatory commitment" is not defined in NRC's regulations, commitments are used in the context of interactions between NRC and licensees for commercial nuclear reactors. The license renewal rule—Title 10, Code of Federal Regulations, Section 54.3 (10 CFR § 54.3)—references commitments in the definition of a "current licensing basis."[1] In February 2000, NRC endorsed Nuclear Energy Institute (NEI)[2] guidance document NEI-99-04, *Guidelines for Managing NRC Commitment Changes,* which the agency found to be an acceptable method for licensees to follow for managing and changing their commitments to NRC. In 2003, NRR released office instruction LIC-105, *Managing Regulatory Commitments Made by Licensees to NRC*,[3] to provide common references for handling commitments.

[1] Per 10 CFR § 54.3 "Definitions," the *current licensing basis* is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect.

[2] NEI is the policy organization for the nuclear technologies industry.

[3] LIC-105 is applicable to NRR staff, and provides them and their stakeholders with a common reference for handling regulatory commitments made by licensees for commercial nuclear reactors to NRC staff.

## Definition of a Commitment

According to NEI-99-04, commitments are docketed, written statements describing a specific action that the licensee has agreed or volunteered to take. Agency and industry guidance documents distinguish between the safety importance and regulatory significance of different types of licensee actions such as obligations and commitments. In summary:

- Obligations are conditions or actions that are legally binding requirements imposed on licensees through applicable rules, regulations, orders, and licenses (includes technical specifications and license conditions).

- Commitments are appropriate for matters that are of significant interest to staff but do not warrant either (1) legally binding requirements, or (2) inclusion in the Updated Final Safety Analysis Reports (UFSAR)[4] or programs subject to a formal regulatory change control mechanism.

Unlike regulatory requirements contained in regulations, technical specifications, licenses, and orders, commitments are neither legally binding nor obligations of a license. According to LIC-105, NRC staff should escalate a commitment into a legally binding obligation only if the staff deems that the commitment is essential for ensuring public health and safety. NRC management asserts that once a commitment is escalated into a requirement, it is no longer a commitment, but rather it becomes a legal obligation and must be converted to an NRC enforceable requirement, such as a condition of the facility operating license.

## Purpose of Commitments

As noted above, commitments are appropriate for matters that are of significant interest to staff, but do not warrant legally binding obligations. According to LIC-105, the regulatory process relies on commitments to, among other things, support the justification for a proposed licensing action or resolution of other regulatory related activities. Commitments

[4] The Final Safety Analysis Report (FSAR) is the principal document upon which the NRC bases its safety evaluation supporting the issuance of an operating license for a nuclear power plant. Changes made after the operating license has been issued are documented in a new document, the UFSAR, which serves as the official source of current plant design and analyses.

often result from a licensing action such as a license amendment, including power uprates, or from a generic communication, such as generic letters and bulletins. Further, LIC-105 states that NRC expects licensees to honor commitments that have a safety or regulatory purpose. Appendix A provides examples of commitments.

NRC expects licensees to honor commitments in good faith; however, noncompliance with a commitment can result in the issuance of a Notice of Deviation. A Notice of Deviation describes a licensee's failure to satisfy a commitment and requests a licensee to provide a written explanation or statement describing corrective steps taken (or planned), the results achieved, and the date when corrective action will be completed.[5] A Notice of Deviation is an administrative mechanism that is less severe than a Notice of Violation,[6] but allows NRC staff to request information from a licensee if the implementation of an action was not consistent with the mutually agreed-upon commitment.

**Licensee Commitment Responsibilities**

According to NEI-99-04, licensees are responsible for creating, implementing, and tracking all commitments made to NRC. As part of their business practices, licensees maintain a commitment management program to track a variety of commitments, including commitments made to NRC[7] and to non-regulatory organizations, as well as corrective actions and self-assessments.

The licensee is entirely responsible for tracking the commitments, and this includes any changes to the commitments and notification to NRC about such changes. NEI-99-04 includes guidance for changing commitments and criteria for determining if and when to inform NRC staff about a change. Although there is no regulatory requirement for such reporting, licensees will periodically report changes in commitments to the NRC via docketed correspondence. NEI-99-04 also provides flowcharts outlining a

[5] According to NRC staff, the agency has not issued any Notices of Deviation to licensees since 2007 for not fulfilling a commitment.

[6] According to NRC's *Enforcement Manual*, a Notice of Violation is a written notice that identifies an NRC requirement and how it was violated.

[7] NRC does not have comprehensive data on the number of commitments made by licensees each year, but licensee staff and NRC project managers estimated that 3 to 10 commitments are created for each plant annually.

commitment management change process to assist licensees in identifying changes that are significant to safety and/or of high regulatory interest that should be communicated to NRC.

**NRC's Commitment Responsibilities**

The primary responsibility for managing commitments within NRC is assigned to DORL project managers, but other NRC staff involved in decisionmaking processes that use commitments include DORL branch chiefs, NRR technical reviewers, and OGC.

DORL project managers are responsible for the general oversight and coordination of NRR activities—including management of commitments—related to the processing of licensing actions, generic issues, or policy issues for a specific licensee. Specifically, as part of NRC's oversight responsibilities, project managers are required every 3 years to audit licensee commitment management programs by assessing the adequacy of licensee implementation of a sample of commitments made to the NRC in past licensing actions.[8] Performance of these audits is not a requirement of NRC's inspection program, which does not assess how well licensees control commitments. Instead, the triennial audit requirement is described in LIC-105.

DORL branch chiefs are expected to ensure that the triennial audits of licensee commitment management programs are performed. In addition, they are expected to ensure that the project managers are appropriately trained and provided with the necessary tools during their reviews of specific licensing actions or other licensing tasks.

It is the role of NRR technical staff to review licensing actions and the supporting documentation—including any applicable commitments—to ensure that appropriate consideration has been given to technical issues. In many cases, project managers will seek subject matter expertise from the technical reviewers during the decisionmaking process.

According to NRR office instructions, NRR staff should coordinate their programmatic efforts with OGC. LIC-100, *Control of Licensing Bases for Operating Reactors;* and NEI-99-04, *Guidelines for Managing NRC Commitment Changes*, states that OGC plays a critical role in defining the

[8] The commitment audit requirement was initiated in 2003. As of February 2010, 65 nuclear power plants had been subject to an initial audit, and 23 plants had been subject to a followup audit.

elements of the licensing bases of nuclear facilities, in defining the appropriate controls for and other attributes of the elements of the licensing bases, and in processing some plant-specific changes to licensing bases information. Further, NRR staff should coordinate their programmatic and plant-specific efforts with OGC to ensure NRC products (e.g., licensing documents) comply with legal requirements.

## II. PURPOSE

The audit objective was to assess the extent to which NRC appropriately and consistently utilizes and manages regulatory commitments for power reactor licensees. Appendix B contains information on the audit scope and methodology.

## III. FINDINGS

Part of NRC's mission is to identify and accomplish those actions that provide the level of nuclear plant performance necessary to ensure adequate protection of public health and safety. A commitment is one tool that NRC uses in the overall licensing process to add flexibility, improve efficiency, and maintain the flow of information between the staff and licensees. The Office of the Inspector General (OIG) identified opportunities for improvement in the following three areas:

- Consistent implementation of commitment management audits.
- Staff understanding of the definition and use of commitments.
- NRC tracking of commitments.

## A. Commitment Management Audits Are Inconsistently Implemented

To achieve NRC's mission of adequately protecting the public health and safety and the environment, NRC programs and processes should be carried out effectively, efficiently, and consistently. However, NRC inconsistently implements the audits of licensee commitment management programs. This is because agency guidance on implementing the triennial audits is incomplete and imprecise. Incomplete and imprecise guidance concerning the conduct of commitment management audits can result in ineffective audits, inefficient use of resources, and the appearance that NRC provides disparate oversight of similarly situated licensees.

### Consistent Implementation of NRC Programs and Processes

To achieve NRC's mission of adequately protecting public health and safety and the environment, NRC programs and processes should be carried out effectively, efficiently, and consistently. Consistent implementation of NRC programs and processes facilitates a consistent regulatory framework for overseeing commercial nuclear power plants. One element of NRC's regulatory oversight process is the management of commitments, which includes conducting triennial audits of licensee commitment management programs. The triennial audits are a primary tool used by NRC for assessing licensee commitment management programs. In accordance with NRC's mission, the triennial audits of licensee commitment management programs should be implemented consistently.

### Commitment Management Audits Are Inconsistently Implemented

NRC's triennial commitment management audits are not consistently implemented. There are disparities in how staff members develop samples of commitments for review and the thoroughness of the audits.

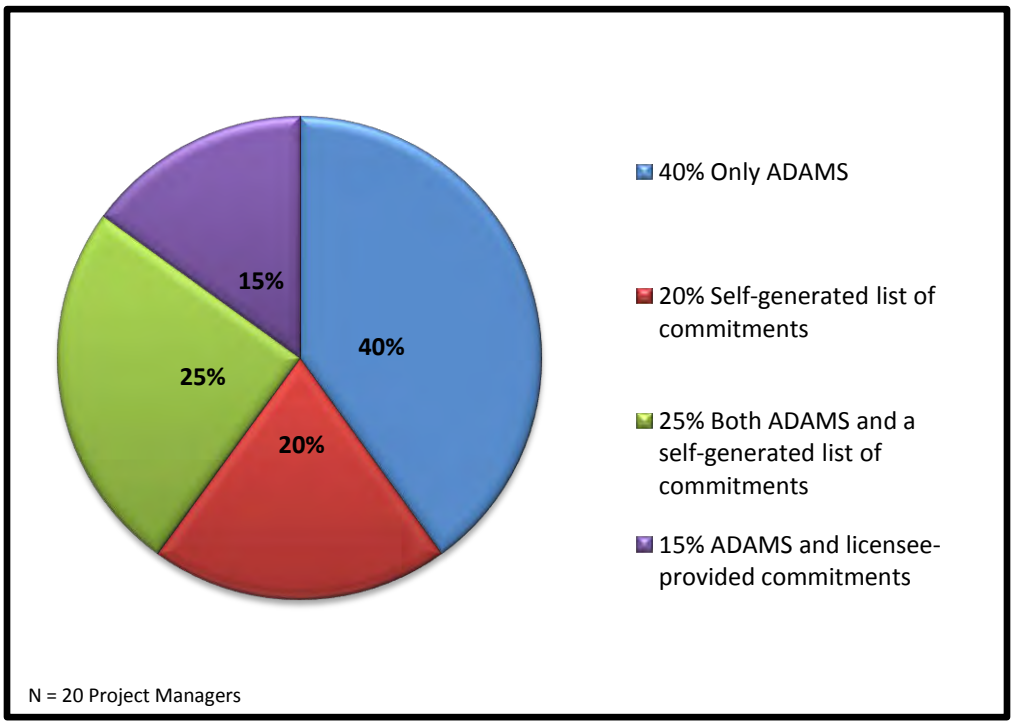
DORL project managers inconsistently identify the universe of commitments eligible for sampling for the triennial audits. Some of the various sources of information that project managers use to identify the universe of commitments include:

- Generic communications.
- Licensee correspondence.
- Prior commitment audit reports.

- UFSAR changes.
- Agencywide Documents Access and Management System (ADAMS).[9]
- NRC amendment logs.
- Licensee commitment tracking systems.
- Licensee corrective action programs.

Most project managers explained that when developing their audit sample, they consult various sources and/or combinations of sources, including ADAMS, their own records, and/or licensee kept records. For example, during the development of a commitment audit sample, some project managers indicated that they depended on a self-generated list of commitments to extract a sample of commitments for the audit, while other project managers said they rely solely upon ADAMS searches as their source of commitments (see Figure 1).

**Figure 1: Percent of Project Managers Indicating Source for Developing Their Audit Sample**




***Source:*** **OIG analysis of interviews with staff.**

[9] ADAMS is the official recordkeeping system through which NRC provides access to libraries or collections of documents related to the agency's regulatory activities.

DORL staff members have varying views on what constitutes a thorough audit, and OIG observed differences in the conduct of the audits. For example, project managers and branch chiefs provided contrasting responses on whether the commitment management audit includes physical verification of commitments. Approximately half of the project managers interviewed said they would not physically or visually inspect the accomplishment of commitments, while all of the branch chiefs that OIG interviewed agreed that project managers should physically or visually inspect the accomplishment of commitments.

Further, OIG observed the performance of several commitment management audits of licensees and noted significant differences in how they were conducted. Two of the audits were performed at licensee nuclear power plant sites while one was performed at NRC headquarters. The two site audits differed markedly from each other in the depth of the review, including the degree to which the project manager reviewed supporting documentation and performed physical verification activities at the nuclear power plant. For example, one of the project managers verified that the licensee had completed a specific commitment, and when asked if further verification was needed, the project manager said that the commitment review stops at the completion of the commitment. However, another project manager reviewed the completion of the commitment and then requested additional supporting documentation that went beyond the documented completion of the specific commitment to ensure the proper changes and procedures had been correctly implemented. The project manager also reviewed associated documents that were directly or indirectly affected by the commitment to ensure that the licensee had made all relevant changes that were impacted by the commitment.

**NRC Guidance Is Incomplete and Imprecise**

Staff interpretations concerning conduct of the triennial commitment management audits vary because agency guidance[10] on conducting the audits is incomplete and imprecise. Specifically, agency guidance on developing a commitment management audit sample does not provide detailed direction on the sources to be used for identifying a universe of commitments. Furthermore, guidance on conducting the audits does not

[10] Although LIC-105 is the agency's primary guidance on managing commitments, other guidance documents such as LIC-100, *Control of Licensing Bases for Operating Reactors,* and LIC-101*, License Amendment Review Procedures,* reference commitments.

articulate the depth-of-review expectations and guidelines for performing the audits.

LIC-105 guidance on developing an audit sample is incomplete because it does not provide detailed direction on sources that should be used to identify the universe of commitments in preparation for an audit. For commitment sample selection, the guidance directs project managers to review commitments specifically made by licensees. The guidance does not offer the specific sources where commitments can be found. Based on this guidance, it is questionable whether project managers consistently draw their samples from an appropriate, representative, or inclusive universe.

Further, the LIC-105 section on conducting triennial commitment management audits is imprecise because it does not clearly articulate the depth-of-review expectations and guidelines for performing the audits. This guidance does not specifically address management expectations of staff to verify that commitments have been appropriately implemented in the plant facility, procedures, program, or other plant documentation. Rather, guidance simply notes that the project manager is responsible for determining the level of physical verification and document review depending on the nature of each commitment. This non-prescriptive approach affords the project manager a degree of flexibility in conducting the audit. However, it does not provide sufficient guidance to ensure that the audits are conducted consistently with regard to thoroughness and level of review.

**Reduced Effectiveness and Efficiency, and the Appearance of Disparate Treatment**

Incomplete and imprecise guidance on the conduct of triennial commitment management audits can result in ineffective audits, inefficient use of resources, and the appearance that NRC provides disparate oversight of similarly situated licensees.

The lack of clear direction from agency guidance contributes to reduced effectiveness of the triennial audits. Unless project managers identify the full universe of commitments for the audit sample and until the guidance provides more clarity with regard to the depth of the audit, the audits may not fully support NRC's objective to determine whether licensees successfully implement their commitments. This may lead to NRC staff

perceptions that the commitment management audits are ineffective. Indeed, some staff articulated to OIG a reluctance to continue performing the audits. Those audits that do not sample a complete universe of commitments or lack the rigor and depth to ensure that commitments were implemented represent an inefficient use of agency resources.

The inconsistencies in the implementation and conduct of the audits also lend an appearance of disparate treatment among licensees. Ideally, NRC should audit two licensees with similar commitments in a similar fashion. However, if two separate project managers reviewed the two licensees with different standards of proof and documentation, the result could be two very different outcomes, giving the appearance of different treatment.

## Recommendations

OIG recommends that the Executive Director for Operations:

1. Revise the section of LIC-105, *Managing Regulatory Commitments Made by Licensees to NRC,* on conducting triennial commitment management audits to include detailed sampling direction, such as a checklist of sources to be used in identifying a universe of commitments from which to sample.

2. Revise LIC-105, *Managing Regulatory Commitments Made by Licensees to NRC,* to include well-defined expectations and guidelines regarding the conduct of commitment management audits. The guidelines should include an expectation that audited commitments are reviewed to ensure that they have been appropriately implemented in the plant facility, procedures, program, or other plant documentations.

## B. Definition and Use of Commitments Are Inconsistently Understood

Agency staff should have a consistent understanding of commitments to perform their work effectively; however, the definition and use of commitments is not consistently understood throughout the agency. This occurs because NRC training on commitments is insufficient. Specifically, training does not effectively address the definition and use of commitments and is not provided to all agency staff involved in reviewing licensee commitments. This could potentially result in the misapplication of commitments by NRC staff.

### Importance of Performance Management

Good performance management practices dictate that agency staff should have a consistent understanding of NRC's regulatory tools and their use, including commitments, so they can perform their work and collectively ensure protection of public health and safety and the environment. Furthermore, the agency's approach to performance management requires staff members to be highly trained in the technical disciplines relating to their duties, the regulatory processes that govern agency actions, and the regulatory principles inherent in making the agency a strong, independent, stable, and predictable regulator. Being a stable and predictable regulator implies that NRC must operate in a consistent regulatory framework. This requires staff associated with a particular regulatory program to have a consistent understanding of that regulatory program and associated policies, including the definitions and the tools used for evaluating and implementing the program.

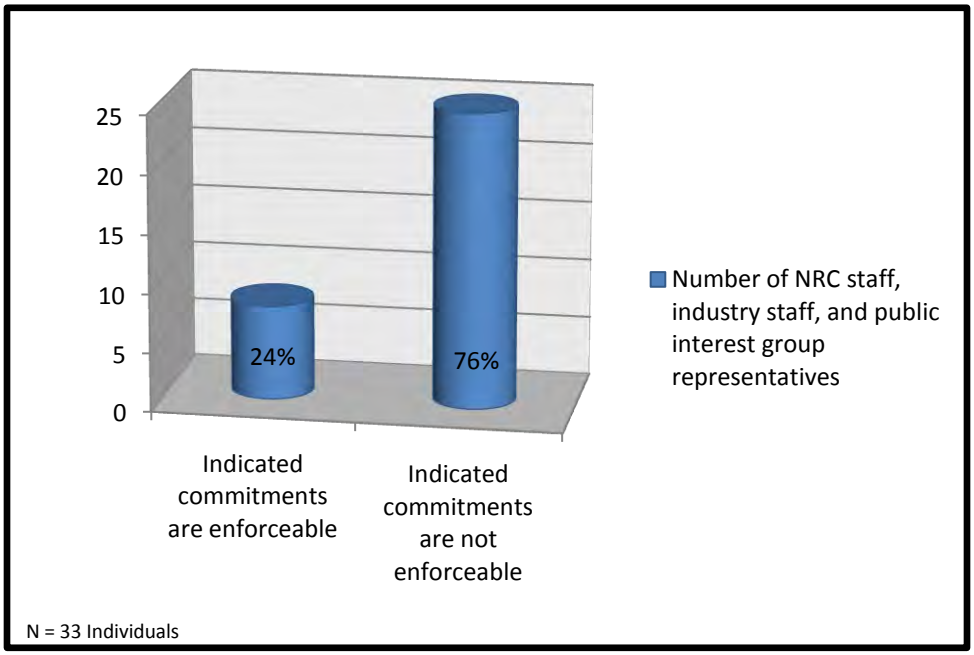
### Definition and Use of Commitments Are Inconsistently Understood

Agency staff, industry, and the public have various views on the definition of a commitment. Furthermore, agency staff expressed conflicting descriptions for the use of commitments.

<u>Commitments Are Variously Defined by Stakeholders</u>

Agency staff, industry personnel, and the public have various views of the definition of a commitment and whether commitments are enforceable. Figure 2 below shows the views expressed by members of these groups when asked whether commitments can be enforced by the NRC.

**Figure 2: Perceptions of Enforceability of Commitments**




***Source:*** **OIG analysis of interview data.**

DORL project managers explained that because commitments are not enforceable,[11] it is their responsibility to assess the level to which NRC staff rely on commitments and to make sure a commitment is the appropriate tool to ensure that the action is completed. Further, NRC staff explained that commitments are not a part of the license and are therefore not legally enforceable.[12] Under that interpretation, NRC could not issue a violation if a licensee fails to fulfill a commitment. However, other agency staff said that NRC could enforce commitments. For example, one branch chief said that a commitment is part of the licensing basis and is therefore enforceable. Other NRC staff members, including project managers, branch chiefs, and an OGC attorney, contended that the agency could take enforcement action if a licensee failed to fulfill the commitment.

Agency and industry staff articulated a hierarchy of commitments that the agency guidance does not specifically address. Some NRC staff have differentiated commitments based on their intended application and NRC's ability to oversee commitments using the terms "big C" commitments and

[11] The term "enforceable" describes a legally binding obligation, such as a condition of a facility's operating license.
[12] According to LIC-105, issues regarding the use of regulatory commitments usually center on the legal standing of the commitment and NRC staffs' ability to enforce the action committed to by a licensee. While licensees are not legally bound to fulfill a commitment, the NRC staff may use the administrative enforcement tool of a Notice of Deviation if a licensee fails to satisfy a commitment.

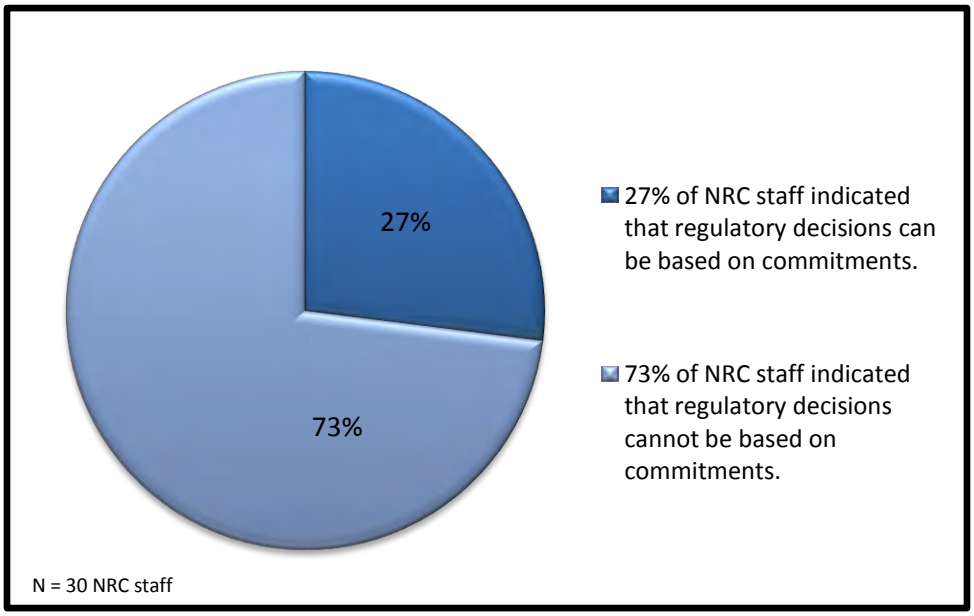
"small c" commitments. Some NRC staff have also used the terms "regulatory commitments" and "commitments" to differentiate between the "big C" and "small c" commitments, respectively. Furthermore, the definitions provided by staff for these different types of commitments were contradictory. For example, one project manager said that "big C" or "regulatory commitments" are safety-significant commitments that could become an obligation or condition of the license, while another project manager along with an OGC staff member explained that a "regulatory commitment" is similar to a "small c" commitment which is a non-enforceable, voluntary agreement between the licensee and NRC. Industry staff also articulated two kinds of commitments—"obligatory commitments" and "non-obligatory commitments"— to differentiate between commitments made in association with a requirement in a regulation and those commitments not explicitly identified in a regulation.

Moreover, there is a difference in the way the public and industry understands a commitment versus NRC staff's view of a commitment. Of 54 NRC staff members interviewed, 28 understood a commitment to be a low-level promise between NRC and the licensees. For example, seven staff members referred to commitments as a "gentleman's agreement" or as the "icing on the cake," essentially providing NRC additional assurance that licensees will take a particular action beyond that specified in the requirements. However, some industry staff and a public interest group see commitments as more formal, enforceable agreements.

Agency Staff Have Conflicting Views on Use of Commitments

Various agency staff involved in handling commitments expressed conflicting descriptions for the use of commitments. Particularly, agency staff members have differing views on whether a regulatory decision (e.g., amendments to licensing documents) can be based on a commitment (see Figure 3). Many NRC staff members believe regulatory decisions should be made without reliance on a commitment and the commitment should serve merely as extra assurance for NRC; however, other staff members believe that licensing actions could be approved with commitments and that, in some cases, NRC could not have approved the licensing action without a commitment.

**Figure 3: NRC Staff Views on Role of Commitments in Regulatory Decisions**




***Source:*** **OIG analysis of interviews with staff.**

Some NRC staff are aware of a regulatory practice that incorporates the content of a commitment into a licensing action implementation statement, while others are unaware of this option. The term "implementation statement" is not defined in the agency guidance for commitments.[13] However, NRC staff members who reported using the implementation statement explained that it requires the licensee to place their commitment(s) into the UFSAR. This makes the commitment subject to the provisions of 10 CFR § 50.59[14] such that changes to the commitment by the licensee would result in a process to determine if prior NRC approval may be required. Additionally, NRC has the opportunity to review changes to the commitment when the UFSAR is updated according

[13] The terms "implementation statement," "implementation clause," and "implementation requirement" were used interchangeably by some NRC staff. Although the term "implementation statement" is not defined in agency commitment guidance, NRC staff explained that they could add an implementation statement or clause to the licensing amendment. This section of the approval letter lists the items to be implemented and the implementation timeframe in conjunction with or prior to the approval of the amendment. The implementation language might state, for example, "the implementation of the amendment shall also include…." Within this "implementation statement" is where a commitment may be inserted and incorporated into the FSAR.

[14] 10 CFR 50.59, *Issuance, Limitations, and Conditions of Licenses and Construction Permits: Changes, Tests and Experiments,* outlines the instances in which a license amendment would or would not be required due to a change in the facility or procedures described in the FSAR (as updated) or a test or experiment not described in the FSAR (as updated).

to 10 CFR § 50.71(e).[15] Therefore, certain commitments that are formally captured and included into the UFSAR through the implementation statement would receive more scrutiny by NRC staff. However, the use of the implementation statement as a tool that allows NRC to have more oversight of selected commitments is not consistently known among NRC staff. Specifically, of DORL staff asked if they had the knowledge and/or understanding of an implementation statement, more than 40 percent said their branch did not use it and they were unaware of such a statement being used by other DORL branches.

**Insufficient Commitment-Specific Training**

Current training on commitments does not sufficiently address the definition and use of commitments, and is not provided to all staff involved in reviewing licensee commitments. Providing commitment-specific training to all applicable NRC staff—including project managers, technical staff, and OGC involved in the formation or revision of reactor licensing actions—helps ensure that staff have the skills, knowledge, and abilities needed to perform their work. DORL is in the process of developing licensing-specific training for project managers that will address the application of commitments; however, this effort has not been addressed by all NRC offices involved in reviewing reactor licensee commitments.

**Misapplication of Commitments**

Until the various understandings of the definition and use of commitments are clarified, NRC risks improper application of commitments. For example, NRC staff may inappropriately rely on a commitment for a licensing decision when an obligation was required. In fact, some NRC staff members said that they would not have approved a particular licensing action without a specific commitment being present. Therefore, lacking a sound understanding of the appropriate application of a commitment, NRC staff may be accepting licensees' proposed commitments in lieu of an appropriate regulatory requirement, such as applicable licensing conditions, orders, rules, or regulations.

[15] 10 CFR § 50.71(e), *Maintenance of records, making of reports,* is the requirement for licensees to update their FSAR that was originally submitted as part of their application for a license. Subsequent revisions must be filed within a period not to exceed 24 months.

**Recommendation**

OIG recommends that the Executive Director for Operations:

3. Develop training that sufficiently addresses the definition and use of commitments and provide it to all agency staff involved in reviewing reactor licensee commitments.

## C. NRC Staff Do Not Systematically Track Commitments

According to Federal regulations for preserving records and NRC guidance on records management, NRC should maintain records that are sufficient to document matters dealt with by NRC, including significant decisions and the decisionmaking process. NRC does not systematically track commitments because the agency does not have an adequate tool for tracking them, in part, because the agency has not identified a need for such a tool. Consequently, NRC cannot completely ensure oversight of commitments, which has implications for the agency's continuing awareness of significant commitments, the effectiveness of the triennial commitment management audits, and institutional knowledge management.

### Preservation of Documents

Federal regulations require NRC to preserve records containing adequate and proper documentation of the functions, decisions, and essential transactions of the agency to ensure that the agency can find records when needed. According to Management Directive (MD) 3.53, *NRC Records Management Program*,[16] the agency should maintain records that are sufficient to document matters dealt with by NRC, including significant decisions and the actions taken to arrive at those decisions. This includes docketed commitments that are considered safety significant[17] and/or relied upon to make regulatory decisions. Documenting commitments that contain information supporting regulatory decisionmaking helps ensure that the agency captures pertinent information and that NRC can be responsive and accountable for its actions in communicating with reactor

[16] MD 3.53 contains procedures, standards, and guidelines for managing NRC's official records in accordance with U.S. National Archives and Records Administration and General Services Administration regulations.

[17] The term "safety significant" refers to a function whose degradation or loss could result in a significant adverse effect on defense-in-depth, safety margin, or risk.

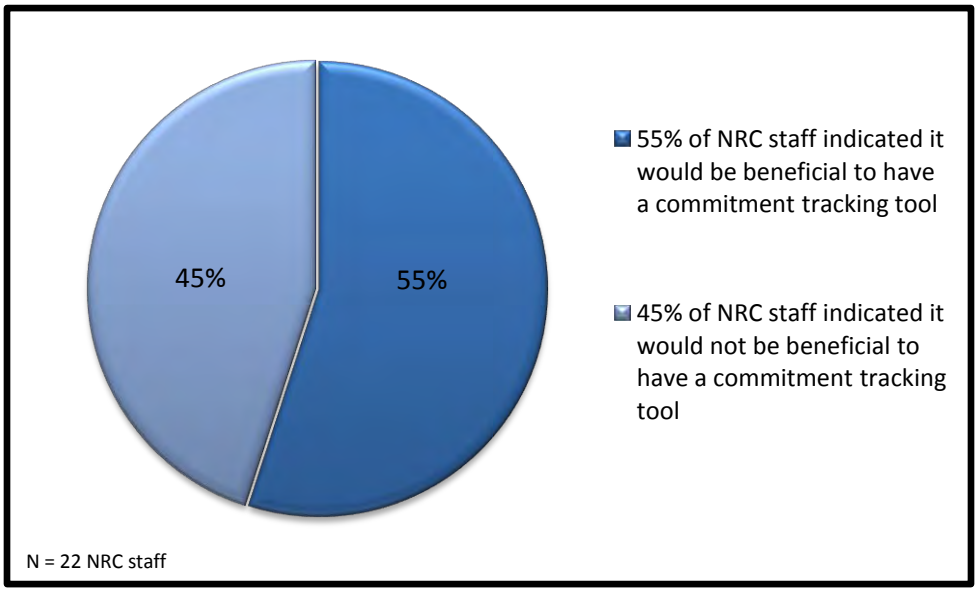
licensees. Moreover, capturing commitments would assist NRC in knowing the universe and status of all commitments.

**NRC Does Not Systematically Track Commitments**

NRC does not systematically track commitments and, consequently, project managers cannot independently generate a list of all commitments related to a specific licensee, even those that were considered by staff to be safety significant and/or integral to approving a proposed licensing action. The agency does not collect commitments into a single document. Rather, commitments are included in various documents submitted by licensees. NRC relies on licensees to track their respective commitments and supply information to NRC on the status of the implementation/ closure of commitments for the purpose of the triennial audit. This is problematic because some staff members view certain commitments as safety significant and/or necessary for approval of proposed licensing actions.

Although project managers are not required to track commitments, OIG learned that some project managers informally track commitments. With no requirement to track commitments, these project managers create their own mechanisms for tracking this information because they seek the ability to independently conduct timely and thorough commitment searches. One experienced project manager explained that maintaining one's own list of commitments provides the opportunity to independently verify that the licensee's commitment-related information is adequately captured, tracked, and managed. Furthermore, the majority of NRC staff interviewed indicated that having a formal tracking tool for licensee commitments would be beneficial (see Figure 4). Many of the staff members who disagreed believe that tracking commitments would not be beneficial did so solely because they felt it would be an administrative burden for project managers.

**Figure 4: NRC Staff View on Whether a Commitment Tracking Tool Would Be Beneficial**




***Source:*** **OIG analysis of interviews with staff.**

## NRC Lacks an Adequate Commitment Tracking Tool

NRC does not have an adequate tool for tracking commitments, in part, because the agency has not identified a need for such a tool. NRC managers said that the agency staff should not rely on commitments for regulatory decisionmaking. However, these managers were unaware that some staff members had used commitments for the approval of licensing actions. Thus, the agency's lack of support for the tracking of commitments has been partly based on an assumption that staff were not using commitments for the purpose of regulatory decisionmaking.

OIG identified some instances of licensee commitments that were safety significant and/or necessary for approval of a proposed licensing action, as follows:

- Commitment A: Staff relied upon a commitment for approval of a licensee's amendment to make a technical specification change regarding reactor power monitoring equipment calibration. NRC's issuance letter stated that the approval of the amendment was based on the commitment. One interviewee—who was the branch chief at the time of the commitment—said, had the commitment not been fulfilled, NRC may have issued an order. Another NRC staff member, a technical reviewer, said that the license amendment

request would not have been accepted without the completion of the commitment.

- Commitment B: After a licensee conducted a power uprate-related evaluation, the licensee made a commitment to operate at a lower power level than allowed by the nuclear power plant license. NRC staff members said that if the commitment had not been completed, it could have adversely impacted safe plant operations. NRC managers involved in the power level approval agreed that they could postulate a safety-related problem had the licensee opted not to implement the commitment. Later, an OGC attorney confirmed that a commitment was not appropriate in this instance; instead, a license amendment should have been used.

- Commitment C: For a requested amendment to extend the allowed out-of-service time for a plant's diesel generators, the staff determined that a commitment was necessary. Two NRC staff members, a technical reviewer and an OGC attorney, said that a commitment to modify a circuit breaker was necessary for the amendment.

OIG did not perform a detailed search and review of commitments to identify commitments, similar to the examples above, that were safety significant and/or necessary for approval of a proposed licensing action. Therefore, it is possible that additional examples exist. The agency also does not know the extent to which such commitments exist because it has not identified commitments that the staff had considered safety significant and/or necessary for approval of a proposed licensing action.

**Agency Cannot Ensure Oversight of Commitments**

Until the agency tracks safety significant commitments, NRC will not be able to ensure oversight of such commitments. Consequently, there are implications for the agency's continuing awareness of significant commitments, the effectiveness of the triennial commitment management audits, and institutional knowledge management. Further, NRC risks erosion of its licensing logic, wherein the agency would rely on non-mandatory commitments in lieu of licensing conditions or obligations for nuclear power plant licensing.

Agency Awareness of Commitments

Without a tracking system, significant commitments could be overlooked or forgotten. For example, a 1979 safety significant commitment resurfaced in 2007 when an NRC inspection found the commitment and issued a Notice of Deviation to the licensee because action to address the commitment had not been completed. If NRC had a process to independently track commitments, the agency would have been able to monitor the implementation of the commitment, and any failure by the licensee to take action would have been identified earlier. Instead, the oversight and tracking of commitment implementation by NRC is ad hoc, making it difficult for the agency and staff members to identify deviations from poorly documented plans. However, effective use of the triennial commitment management audits to identify potential inappropriately-applied commitments and agency training on the proper implementation of commitments, once implemented and reviewed by OIG, may obviate the need for a tracking system.

Impact on Triennial Commitment Management Audits

NRC risks conducting ineffective triennial commitment management audits because significant commitments may not be included in the commitment management audit samples. OIG learned during an audit it observed that NRC missed commitments made during an entire year between NRC's initial and first followup audit of a licensee's commitment management program. OIG notified the project manager of the missed timeframe, and the project manager stated a belief that it was unnecessary to increase the audit sample to include the missed year's commitments because the licensee performs its own internal audits.

Impact on Institutional Knowledge Management

Employee attrition could potentially result in the agency's loss of undocumented information, particularly in those instances where some project managers have developed their own commitment tracking systems. Commitment related information to support future projects or regulatory decisionmaking may not be available for germane agency staff if the agency does not formally capture the information. The commitment management audits should identify and correct situations where commitments were used inappropriately. Properly documenting relevant information is a critical aspect of effective oversight and demonstrates that

NRC operates with due care and can be accountable for its oversight of commitments.

Risks to NRC's Licensing Logic

NRC risks not following its established licensing logic, leaving the agency in a potential position whereby the licensing of nuclear power plants depends on non-mandatory commitments. The licenses for operating nuclear power plants include requirements that ensure that the functional capability or performance levels of equipment required for safe operation of the facility are maintained. The requirements in licenses are mandatory and require compliance by the licensee. If NRC allows reliance on commitments for the approval of license amendments, it risks making the basis of safe operation reliant on actions that are not required.

## Recommendations

OIG recommends that the Executive Director for Operations:

4. Identify actions to determine the extent to which commitments that are considered safety significant and/or necessary for approval of proposed licensing actions exists. This could be accomplished by either: (1) NRR project managers identifying any such commitments as part of the triennial commitment management audits, or (2) conducting a review of all existing commitments and identifying any inappropriately applied commitments. Any such commitments should be identified to NRC management for appropriate action.

5. Depending on the outcome of the efforts to meet recommendation 4, develop and utilize a tool for systematically tracking the status of commitments that are deemed safety significant and/or necessary for approval of proposed licensing actions.

# IV. CONCLUSION

NRC commitments are a valuable regulatory tool that add flexibility to the regulatory review framework. They also play a key role in facilitating the agency's decisionmaking process on matters that can be highly safety significant. Specifically, they provide additional assurance to the agency that a licensee action will not adversely affect the safe operation of the plant. Therefore, it is essential that all agency staff who work with commitments clearly understand the appropriate application and role of commitments to facilitate their consistent use. However, not all NRC staff understand the appropriate use of commitments. By enhancing agency guidance and training on the role and use of commitments, as well as requiring routine review and capture of commitments pertaining to safety-significant decisions, the agency can further strengthen its pledge to promote the safe operation of the nation's power reactors.

# V. CONSOLIDATED LIST OF RECOMMENDATIONS

OIG recommends that the Executive Director for Operations:

1. Revise the section of LIC-105, *Managing Regulatory Commitments Made by Licensees to NRC,* on conducting triennial commitment management audits to include detailed sampling direction, such as a checklist of sources to be used in identifying a universe of commitments from which to sample.

2. Revise LIC-105, *Managing Regulatory Commitments Made by Licensees to NRC,* to include well-defined expectations and guidelines regarding the conduct of commitment management audits. The guidelines should include an expectation that audited commitments are reviewed to ensure that they have been appropriately implemented in the plant facility, procedures, program, or other plant documentations.

3. Develop training that sufficiently addresses the definition and use of commitments and provide it to all agency staff involved in reviewing reactor licensee commitments.

4. Identify actions to determine the extent to which commitments that are considered safety significant and/or necessary for approval of

proposed licensing actions exists. This could be accomplished by either: (1) NRR project managers identifying any such commitments as part of the triennial commitment management audits, or (2) conducting a review of all existing commitments and identifying any inappropriately applied commitments. Any such commitments should be identified to NRC management for appropriate action.

5. Depending on the outcome of the efforts to meet recommendation 4, develop and utilize a tool for systematically tracking the status of commitments that are deemed safety significant and/or necessary for approval of proposed licensing actions.

## VI. AGENCY COMMENTS

On August 9, 2011, OIG issued the discussion draft of this report to the Executive Director for Operations. OIG met with NRC management officials and staff on August 12, 2011, at which time the agency provided informal comments to the draft report. OIG subsequently met with agency management and staff during an August 23, 2011, exit conference to discuss agency informal comments that OIG incorporated into the draft report as appropriate. At this meeting, agency management provided supplemental information that has also been incorporated into this report as appropriate. NRC management and staff reviewed the revised draft report and generally agreed with the recommendations in this report. Furthermore, the agency opted not to provide formal comments for inclusion in this report.

**Appendix A**

# EXAMPLES OF COMMITMENTS

Commitments are generated from different sources, including license amendments, notices of enforcement discretion, generic letters, and other operational and licensing documents. However, the commitments are documented as written communication from the licensee to NRC. The following examples illustrate some of the sources and types of commitments utilized by the industry and NRC.

Example 1: Commitment to upgrade a spent fuel pool crane

> All heavy load lifts in or around the spent fuel pool made using the upgraded Auxiliary Building crane lifting system will meet the guidance in NUREG-0612.

This commitment was made in support of a license amendment related to the plant's spent fuel pool crane. In this case, the licensee committed to limit use of the crane so that objects above a specific weight would not be lifted unless the crane was upgraded.

Example 2: Commitment to maintain a minimum amount of fuel oil available

> The licensee commits to administratively control the amount of fuel oil in each fuel oil storage tank such that a minimum usable amount of 25,000 gallons (including the day tanks) is available to supply each EDG [emergency diesel generator] (without reliance on a portable transfer pump), for the duration of the enforcement discretion.

In this case, a licensee made a commitment in support of the licensee receiving approval for temporary enforcement discretion for a requirement related to an emergency diesel generator. The licensee committed to maintain an amount of fuel in the plant's fuel oil storage tanks that was greater than the minimum amount normally required.

Example 3: Commitment to modify the containment emergency sump of a nuclear power plant

> Installation of Unit 1 and Unit 2 new post loss of coolant accident containment sump recirculation screens, completion of required modifications and implementation of required procedural changes.

A licensee made this commitment to the NRC in response to an NRC Generic Letter. NRC originally sent the Generic Letter to licensees of pressurized water reactors to communicate a generic concern with their containment emergency sumps. In response to this concern, this licensee committed to make modifications to its reactors' sumps recirculation screens.

**Appendix B**

# SCOPE AND METHODOLOGY

The audit objective was to assess the extent to which NRC appropriately and consistently utilizes and manages regulatory commitments for power reactor licensees. The audit focused on reviewing the oversight of commitments through an examination of documents, interviews, and observations.

OIG reviewed relevant Federal regulations regarding the management and use of commitments, including 10 CFR § 54.3, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*, and 10 CFR § 50.59, *Issuance, Limitations, and Conditions of Licenses and Construction Permits: Changes, Tests and Experiments.* OIG also reviewed agency and industry guidance, including LIC-105, *Managing Regulatory Commitments Made by Licensees to the NRC*; LIC-100, *Control of Licensing Bases for Operating Reactors;* and NEI-99-04, *Guidelines for Managing NRC Commitment Changes.* OIG reviewed generic communication documents as well as licensing documents such as license amendments. Furthermore, OIG reviewed NRC inspection procedures, SECY papers, office handbooks, and all (88) NRC commitment audit reports published between January 2004 and February 2009.

OIG interviewed NRC staff who participated in activities related to the management of commitments. These interviews included resident inspectors, OGC staff, NRC technical reviewers, DORL project managers, DORL staff, and agency managers. In all, OIG conducted interviews with 54 NRC staff members to obtain staff insights into the oversight of licensees' commitments and commitment management programs.

OIG also conducted interviews with industry representatives, a public interest group representative, and licensee staff. The audit team also observed three audits of licensee's commitment management programs performed by DORL project managers.

This performance audit was conducted at NRC headquarters (Rockville, MD) and selected commitment audit sites in Regions II and III, from October 2010 through May 2011, in accordance with generally accepted Government auditing standards. Those standards require that the audit is planned and performed with the objective of obtaining sufficient,

appropriate evidence to provide a reasonable basis for any findings and conclusions based on the stated audit objective. OIG believes that the evidence obtained provides a reasonable basis for the report findings and conclusions based on the audit objective. Internal controls related to the audit objective were reviewed and analyzed. Throughout the audit, auditors were aware of the possibility or existence of fraud, waste, or misuse in the program.

The audit work was conducted by R.K. Wild, Team Leader; Kevin Nietmann, Senior Technical Advisor; Jaclyn Storch, Audit Manager; Andrea Ferkile, Senior Management Analyst; Joseph Capuano, Auditor; and Dana Furstenau, Student Management Analyst.