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"Performance Assessment for Radioactive Waste Management at Sandia National Laboratories: A 30-year History"

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ABSTRACT

Over the past three decades, Sandia National Laboratories has developed and applied a performance assessment (PA) methodology that has informed key decisions concerning radioactive waste management. This experience includes not only the WIPP and Yucca Mountain projects, but also the initial development and demonstration of the U.S. Nuclear Regulatory Commission's initial PA capabilities for both high-level and low-level wastes, the subseabed disposal program, PAs for wastes stored at the Idaho National Engineering Laboratory, and PAs for greater confinement disposal (GCD) boreholes at the Nevada Test Site, as well as multiple international collaborations. These efforts have produced a generic PA methodology for the evaluation of total waste management systems that has gained wide acceptance within the international community. More importantly, this methodology has been used as an effective management tool to evaluate different disposal designs and sites; inform development of regulatory requirements; identify, prioritize and guide research aimed at reducing uncertainties for objective estimations of risk; and support safety assessments. This PA methodology could be adapted to evaluate analyses of different strategies and options that might be proposed to manage the back-end of the nuclear fuel cycle.

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INTRODUCTION

Over the past three decades, Sandia National Laboratories (SNL) has developed and applied a performance assessment (PA) methodology that has informed key decisions concerning radioactive waste management both in the United States (U.S.) and internationally. A PA is a probabilistic risk analysis of a radioactive waste disposal facility conducted to demonstrate a reasonable expectation that performance objectives established for the long-term protection of human health and the environment will not be exceeded following permanent closure of the facility. It has historically been used to assess the long-term performance of nuclear waste repositories, but this approach has also been applied to near-surface landfills, waste-disposal sites, and high-level radioactive waste tanks.

This paper will summarize the work conducted by SNL in the following PA applications:

- Siting, characterization, and PA for the international program on subseabed disposal [1,2,3,4,5,6,7,8];
- Development and demonstration of spent nuclear fuel (SNF) and high-level waste (HLW) PA methodology for the U.S. Nuclear Regulatory Commission (NRC) [9,10];
- Development and demonstration of low-level waste (LLW) PA for the NRC [11];
- Evaluation of two generic geologic repositories for disposal of HLW and SNF stored at Idaho National Engineering Laboratory (INEL) [12];
- Development and implementation of the Waste Isolation Pilot Plant (WIPP) PA including two recertification iterations [13,14,15,16,17,18];
- Environmental assessment of proposed repository at Yucca Mountain during HLW site selection [19];
- Development and implementation of the Yucca Mountain Project PA [20,21,22,23,24]; and
- Development and implementation of Greater Confinement Disposal (GCD) PA for special-case wastes [25,26,27] and a preliminary PA for disposal of vitrified Fernald waste [28].

These efforts have produced a generic PA methodology for the evaluation of total waste management systems that has gained wide acceptance (see Figure 1) within the international community. More importantly, this methodology has been used as an effective management tool to evaluate different disposal designs and sites; inform development of regulatory requirements; identify, prioritize and guide research aimed at reducing uncertainties for objective estimations of risk; and support safety assessments. With modest variation, each PA follows the same steps outlined in Figure 1. The approach consists of characterizing the overall system (i.e., waste, facility, and site), scenario selection and screening, building the system model (both conceptual and numerical models), consequence modeling (e.g., source term, groundwater flow, radionuclide transport, biosphere transport, and health effects), and evaluating system performance including uncertainty and sensitivity analysis. The results inform the science and testing program and allow prioritization of future program needs. SNL's PA methodology is iterative, allowing new information and data to be incorporated as it becomes available, and enabling efforts to be focused on what is most important to performance measures, such as dose.

This paper discusses the historical application of the PA methodology and concludes with a brief discussion on the potential application of this methodology to the analysis of different strategies and options that might be proposed for managing the back-end of the nuclear fuel cycle in the U.S. Herein we refer to the back-end of the fuel cycle as those processes necessary to safely manage nuclear wastes, such as storage, transportation, and disposal, and nuclear waste includes SNF, HLW, LLW, and Greater-Than-Class-C (GTCC) waste.

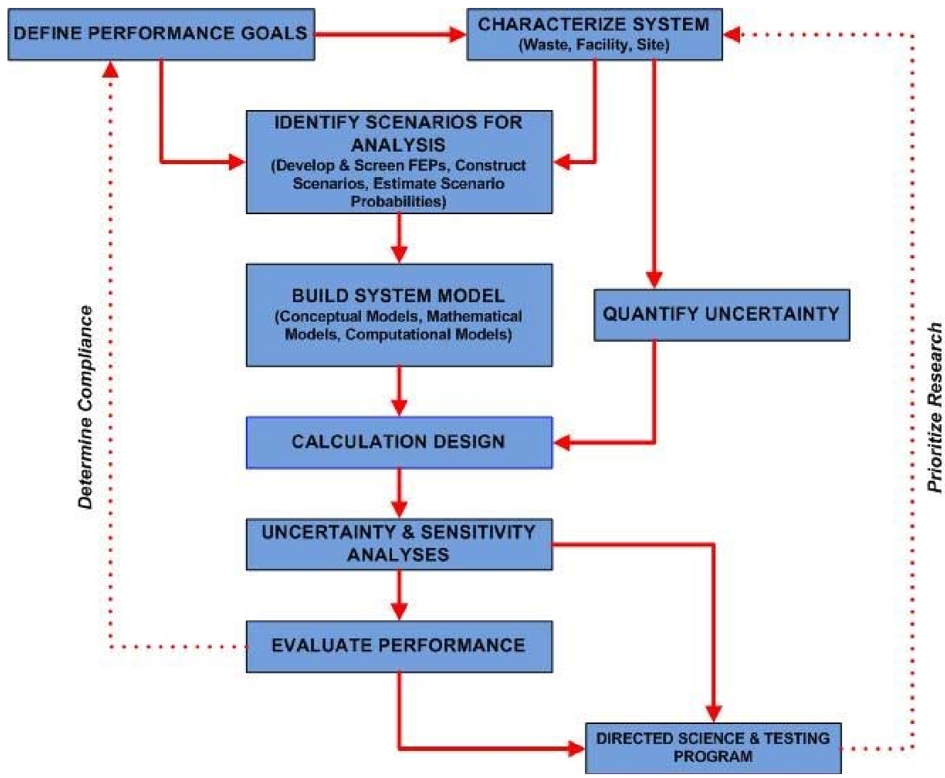

Figure 1. SNL's PA methodology.

HISTORICAL APPLICATION OF THE PA METHODOLOGY

International Subseabed Disposal Program

The U.S. Subseabed Disposal Project (SDP) was part of the international Seabed Working Group (SWG) of the Organisation for Economic Co-Operation and Development/Nuclear Energy Agency, and operated between 1976 and 1987 [1] . The SDP/SWG examined the feasibility of disposal in the sub-bottom clay sediments of abyssal plains under approximately 6,000 meters (m) of water. Site selection focused on three study areas where geologic characterization was performed, one eastern Atlantic site on the Madeira Abyssal Plain, one western Atlantic site on the southern Nares Abyssal Plain, and one north western Pacific site east of the Shatsky Rise [2]. Ocean circulation models were assembled using a nested approach from bottom sources in the deep bottom boundary layer through regional-scale open-ocean models to ocean-basin scales to describe oceanic dispersion of radionuclides [3]. Field studies provided physical data that was synthesized and assimilated into ocean circulation models. Simplified models for PA purposes were abstracted and focused on bottom sources in the Southern Nares Abyssal Plain as a reference location for risk assessment calculations.

Waste canisters were to be emplaced in the sub-bottom clays via free-fall penetrators reaching a nominal depth of 50 m below the sediment-water interface [4]. Laboratory investigations of radionuclide migration through the sub-bottom deep sea sediments comprised of sorption and diffusion experiments that provided data for risk assessment calculations [5]. Studies of processes near the buried waste canisters included thermal, induced pore water flow, canister corrosion, waste form degradation, sea-water/sediment interaction experiments, and modeling also provided abstractions and data to PA [6]. Extensive field and modeling studies of deep-ocean biological processes were performed to understand the role of living organisms and the carbon cycle in dispersing radionuclides; seafood pathways to humans and impacts on the ocean ecosystems provided data and model abstractions to PA, also [7]. The above models and data were used in early implementations of the PA methodology shown in Figure 1 [8].

The steps in the PA were iterative, and were improved and updated as the design developed and as the models and database became more complete. The PA system component models were exact analytical solutions or process model abstractions, each with physical and chemical limitations. Sensitivity analyses revealed a low sensitivity to most natural and design parameters. Burial depth and vertical pore water velocity in the sub-bottom sediments were the most important parameters [8].

The maximum estimated individual dose, as well as the maximum estimated collective dose, would occur approximately 100,000 years after disposal and were 6 and 8 orders of magnitude, respectively, below average background levels. The PA results aided the planning of the research program, site selection criteria, and performing the functional analyses for design of the penetrator and emplacement ship [1]. A system engineering approach was applied based generally on Figure 1. Thus, the SDP as part of the SWG was the first implementation of the PA methodology for the DOE performed by SNL.

Early Work for NRC

Generic HLW Repository in Bedded Salt

Around the same time that the subseabed program was getting started, the NRC instituted a program with SNL to develop a risk assessment methodology for use in evaluating potential disposal sites for radioactive waste. The ensuing methodology followed the same steps outlined Figure 1, and the first demonstration of that methodology was for a hypothetical HLW repository within a bedded salt formation [9], an unfractured saturated porous medium. Models were developed to evaluate the regional hydrology, waste/host rock interactions, groundwater flow and solute transport, biosphere radionuclide transport, and dosimetry and health effects.

The demonstration included the evaluation of 12 different scenarios, including an undisturbed (or base-case) scenario, as well as multiple disturbed repository scenarios involving various combinations of boreholes and groundwater withdrawal wells. Three scenarios were deemed representative of all 12 and received a full consequence and sensitivity analysis; thus, scenario screening was performed. Random samples of 33 uncertain parameters were obtained from their respective probability density functions (PDFs) using Latin hypercube sampling (LHS), and 105 input vectors were constructed for each scenario.

The results were compared to the then working draft of the U.S. Environmental Protection Agency (EPA) standard contained in 40 CFR 191, Subpart B, regulating the deep geologic disposal of radioactive waste. This standard requires that results be expressed in terms of probabilities of release and integrated radionuclide releases over 10,000 years. In order to compare results to this risk-based standard, complementary cumulative distribution functions (CCDFs) are generated; a CCDF indicates the probability of exceeding various levels of cumulative release. The results for the bedded salt PA demonstration violated the draft EPA standard, and owing to the structure of the calculation, it was possible to identify the individual vectors that caused the limit to be exceeded [9]. The PA successfully demonstrated the probabilistic risk assessment methodology for a deep geologic repository.

Generic HLW Repository in Basalt

Building upon the earlier work in bedded salt, SNL then extended its PA methodology to assess the long-term performance of a HLW repository in a saturated fractured basalt formation [10]. The same general methodology was followed with the main difference being the conceptualization of groundwater flow and transport. Bedded salt was assumed to behave as a porous medium whereas basalt, which is dominated by fractures, required a different set of flow and transport models and codes to assess consequences.

The reference site for the PA was the Columbia Intermontane Province near Hanford, Washington, in the northwestern U.S., although the geohydrology was greatly simplified for this demonstration. Three performance measures were examined: 1. integrated discharge of each of the 30 radionuclides in the transport model, 2. normalized EPA sum of radionuclide discharge to the accessible environment in 10,000 years (i.e., the containment requirements in 40 CFR Part 191), and 3. groundwater travel time from the edge of the repository to the accessible environment (10 CFR Part 60).

A total of 318 credible scenarios were developed, which were then screened based on probability of occurrence. Scenarios with similar consequences were grouped and only the scenario with the highest frequency of occurrence in each group was analyzed. This resulted in seven scenarios for evaluation: 1. heat-loading effects caused by heat released from the waste, 2. mechanical loading effects caused by an advancing glacier, 3. normal or pre-emplacement groundwater flow in the region (base-case), 4. groundwater pumping from confined aquifers, 5.change of river location near the repository, 6. drilling a borehole through the repository, and 7.generation of a fault through the repository. Uncertainty analysis was based on Monte Carlo simulations. Seventy input vectors were constructed based on the LHS sampling of 57 uncertain parameters. Sensitivity analysis was performed on the results of the base-case scenario (pre-emplacement groundwater flow) [10].

The results of the basalt demonstration showed that when matrix diffusion was small, the radionuclides were essentially unretarded. Hence, it was possible to violate the standard in 40 CFR Part 191. In addition, simulations beyond 10,000 years exhibited high releases due to dispersion effects. The demonstration results led the researchers to conclude that the PA methodology: 1. could be applied to other saturated fractured media; 2. was capable of assessing the performance criteria in 10 CFR Part 60 and 40 CFR Part 191; and 3. it effectively incorporated uncertainty/sensitivity into the analysis. The demonstration also produced a set of thoroughly documented codes and procedures [10].

Generic LLW PA

A generic PA methodology was then developed for the NRC for evaluating license applications for LLW disposal facilities [11]. Low-level waste facilities are generally shallow land burial sites, such as trenches or boreholes, as opposed to deep geologic disposal. The methodology followed the same steps described previously, and it was demonstrated using a simple conceptual model involving land burial of C-14 in a shallow trench. The methodology provided NRC with a tool for performing confirmatory analyses to evaluate whether a licensee's analyses and assumptions were reasonable and to compare calculated estimates of performance against the performance objectives in 10 CFR Part 61.41, Licensing Requirements for Land Disposal of Radioactive Waste.

The PA considered dose to individuals from off-site releases under normal conditions, as well as on-site doses to inadvertent intruders. The models included: 1. groundwater flow, 2. source term, 3. ground-water transport, 4. surface water transport, 5. air transport, and 6. pathways and dosimetry. The PA produced a series of dose histories for each radionuclide of importance. The contribution of each individual radionuclide to the dose was summed to produce the total estimated dose, which could then be compared to 10 CFR 61.41. The PA successfully demonstrated to the NRC that the methodology was a valuable tool in evaluating LLW facility license applications.

Deep Geologic Repository PA for U.S. Department of Energy (DOE)

Generic Salt and Generic Granite Repositories for DOE-Owned Waste

SNL evaluated two generic fully-saturated geologic repositories, a bedded salt and a partially fractured granitic rock, for the geologic disposal of HLW and SNF being stored at INL [12].

Following the same PA methodology, the results showed that, for the set of generic assumptions made for this exercise, both repository types had the potential to meet the performance criteria specified in 40 CFR Part 191. The salt repository met the criteria even without a moderately corrosion-resistant canister or chemically-adsorptive backfill. In the salt repository, differences among the individual waste forms were less pronounced than in the granite repository and the differences nearly disappeared when the waste forms were combined as a disposal group with vitrified HLW. The PA results provided INL decision-makers with additional detail that could help guide them in preparing their stored wastes for permanent disposal [12].

Bedded Salt (WIPP)

The WIPP site located approximately 42 km due east of Carlsbad, New Mexico, has been developed by the DOE for the deep geologic disposal of defense-related transuranic (TRU) waste. The facility is located 655 m underground within a geologically stable salt formation known as the Salado. Following an extensive site characterization effort, SNL conducted a demonstration PA in 1989 and three full PAs in 1990, 1991, and 1992 [13,14,15] (see Table 1). Results of the 1992 WIPP PA led DOE to conclude that the site was suitable for the disposal of TRU waste, and DOE proceeded on a path to certification, culminating in the submittal of the Compliance Certification Application (CCA) to the EPA in October 1996 [16]. WIPP became the first deep geologic repository certified in the U.S. to permanently dispose of TRU waste generated from the research and production of nuclear weapons, and the site received its first shipment of TRU waste on March 26, 1999.

Table 1. History of DOE WIPP PAs [17,18,29,30]

PA Iteration	Purpose and Summary of Key Results
1989 PA Demonstration	-Demonstration of PA approach. -No release without human intrusion. -27 sampled parameters.
1990 first full PA	-First full PA using CAMCON modeling system. -40 sampled parameters. -Included both scenario and parameter uncertainty. -Direct releases at the surface from drilling intrusion were most important release pathway.
1991 second PA	-Major models linked in PA. -46 sampled parameters. -Direct releases at the surface from drilling intrusion were most important release pathway.
1992 third PA	-Models and data refined. -Uncertainty in transmissivity fields refined. -49 sampled parameters. - Direct releases at the surface from drilling intrusion were most important release pathway.
1996 WIPP CCA	-Included MgO backfill, potash mining and subsidence scenario. -Higher intrusion rate. -57 sampled parameters. - Direct releases at the surface from drilling intrusion were most important release pathway. -Significantly smaller estimated release than release limits.
1997 EPA PA Verification Test (PAVT)	-EPA-selected values for 26 parameters and EPA-selected model assumptions. -Confirmed CCA results.
Compliance Recertification Application (CRA)-2004	-Alternative panel closure (Option D) explicitly included. -Three of 24 conceptual models modified: disposal system geometry, repository fluid flow, and disturbed rock zone. -New model for the spalling component of direct releases during drilling. -Significantly smaller estimated release than release limits.
CRA-2009	-Parameter updates, code improvements, and corrections. -Upgrades to the computational platform. -Significantly smaller estimated release than release limits.

The WIPP PAs have followed the same general methodology over the past 20 years with continual improvements and refinements of the conceptual models, the input parameters, and computer codes. The methodology examines potential release scenarios, quantifies their likelihoods, estimates potential releases to the accessible environment, and evaluates the potential consequences. The WIPP PA system ties together process models for initial activity and subsequent radioactive decay of multiple waste streams, gas generation due to metal container corrosion and microbial degradation of organic waste components, disposal room closure, brine and gas flow within the repository, actinide solubility and mobilization in brines, direct releases (contaminated solids and brine) to the surface from drilling intrusions and long-term releases due to far-field transport of contaminated groundwater.

In the CCA PA [16], 57 uncertain parameters were sampled using LHS and 100 vectors were assembled. Random sampling of the occurrence of possible future events generated the possible futures that give us the CCDF. The PA results showed that radionuclide release to the accessible environment boundary is negligible for the undisturbed scenario and has no impact on compliance. For the disturbed scenario, there are four potential release mechanisms: cuttings and cavings, spallings, direct brine release at the surface during drilling, and groundwater releases following groundwater transport. Cuttings and cavings are the most significant contributors to the mean CCDF while spallings make a small contribution and direct brine releases are less important. The most significant parameters are microbial degradation of cellulose, shear strength of waste, corrosion rate for steel, waste particle diameter, initial value for halite permeability, borehole permeability, increase in brine saturation of waste due to capillary force, and anhydrite permeability. The CCA results demonstrated with greater than 95% confidence that the overall mean CCDF is in compliance with the containment requirements contained in 40 CFR 191.13 [30].

EPA regulations also require recertification every five years following the first receipt of waste. In the first [17] and second [18] recertifications of the WIPP (see also Table 1), the PA results continue to show that the estimated releases are well below release limits (Figure 2) as was shown in the CCA PA.

Figure 2. WIPP PA Results for the Second Recertification (CRA-2009) [18].

<u>Welded Tuff (Yucca Mountain)</u>

Yucca Mountain, located approximately 161 km northwest of Las Vegas, Nevada, within the boundaries of the Nevada Test Site (NTS), was proposed to be the nation’s first repository for the disposal of military and civilian SNF and HLW. On June 3, 2008, the DOE Office of Civilian Radioactive Waste Management (OCRWM) submitted its license application [20] to the NRC for authorization to construct the Yucca Mountain repository. A critical component of that license application was the Yucca Mountain total system performance assessment (TSPA) led by Sandia National Laboratories.

In addition to the 2008 TSPA that OCRWM submitted to the NRC for licensing (TSPA-LA) [20], SNL was involved in multiple iterations of the Yucca Mountain TSPA over the last 20 years (Table 2) including the environmental assessment for Yucca Mountain [19], the initial TSPA [21], the second iteration TSPA [22], the viability assessment [23], and the site recommendation [24], as well as several iterations between 1996 and 1998. However, it should be noted that SNL did not lead those efforts.

Table 2. History of DOE Yucca Mountain TSPAs.

TSPA Iteration	Purpose and Summary of Key Results
TSPA-1991	-Demonstration of TSPA approach. -Volcanism identified importance of uncertainty in unsaturated zone (UZ) flow paths.
TSPA-1993	-Improved models for UZ and saturated zone (SZ). -Evaluated alternative models. -Included early models for thermal processes and engineered barrier system (EBS). - Identified importance of uncertainty in thermal hydrology, UZ flow, and corrosion of engineered materials.
TSPA-1995	-Incorporated new science and design; evaluated alternative models of flow about package. -Identified importance of process models to WP degradation, seepage, UZ and SZ transport.
1998 TSPA-VA	-Supported the 1998 Viability Assessment. -Models based on best current information. -Ranked importance of uncertainty in each of the major components for 10,000, 100,000 and 1,000,000 years. -Emphasis on seepage, water chemistry, corrosion, and SZ.
1999 License Application Design Selection (LADS)	-TSPA tools used to evaluate relative merits of design alternatives. -Demonstrated that multiple designs were viable for long-term performance.
2000 TSPA for Site Recommendation (TSPA-SR)	-Modeling system used fully qualified inputs. -Conservative approach adapted for some components. -Importance of volcanism identified. -Conservative treatments of uncertainty complicated realistic understanding.
FY 2001 Supplemental Science and Performance Analyses (SSPA)	-Included more realistic treatment of uncertainty. -Incorporated new information since TSPA-SR. -Confirmed potential suitability. -Confirmed importance of volcanism and EBS performance for 10,000 years. -Insights into EBS and natural system effects on peak dose.
2001 TSPA for Final Environmental Impact Statement (FEIS)	-Updated SSPA to include new information, and revised regulatory boundary.
2008 TSPA-LA	-Added seismic disruptive event. -Models updated to current information.

The YMP TSPA model is a system-level model that integrates submodels for each of the various components of the natural and engineered barriers. The TSPA model relies on simplifications, or abstractions, of some of the major processes due to the complexity of those processes and the large number of system-level simulations required for the Monte Carlo uncertainty analysis. For the 2008 TSPA-LA [20], a total of 374 FEPs were identified and 222 were excluded, leaving 152 FEPs in the analysis. Four discrete scenario classes were analyzed probabilistically including: 1. an early failure scenario class, in which one or more waste packages or overlying drip shields fails prematurely due to undetected manufacturing or emplacement defects, 2. an igneous disruption scenario class in which a volcanic event causes magma to intersect the emplacement region, with or without an accompanying eruption, 3. a seismic disruption scenario class, in which ground motion or fault displacement damages waste packages and drip shields, and 4. a nominal scenario class in which none of these three types of events occurs.

Each event-based scenario class was subdivided into separate modeling cases to simulate the consequences of specific events. The total mean annual dose for 10,000 years was developed by summing the mean annual doses for each modeling case. The TSPA-LA results [20] were well below the regulatory limits for both the NRC and EPA rules for 10,000 years (Figure 3) and 1,000,000 years, 10 CFR Part 63 and 10 CFR Part 197, respectively. The estimated 10,000-yr maximum mean annual dose was 0.0024 mSv (0.24 mrem) and the 10,000-yr standard is 0.15 mSv (15 mrem). The TSPA-LA results indicated that the largest contributors to the estimated maximum mean annual dose came from the igneous intrusion and seismic ground motion scenario classes, considering the probability of occurrence of these events. The primary release mechanism late in the million-year period was nominal corrosion processes that lead to degradation and failure of the waste packages.

Although the current Administration has decided that Yucca Mountain is no longer an option for the long-term management of nuclear waste in the U.S., acceptance of the Yucca Mountain License Application for docketing by the NRC represents a significant milestone. Thus, the TSPA-LA is a significant piece of work in that regard.

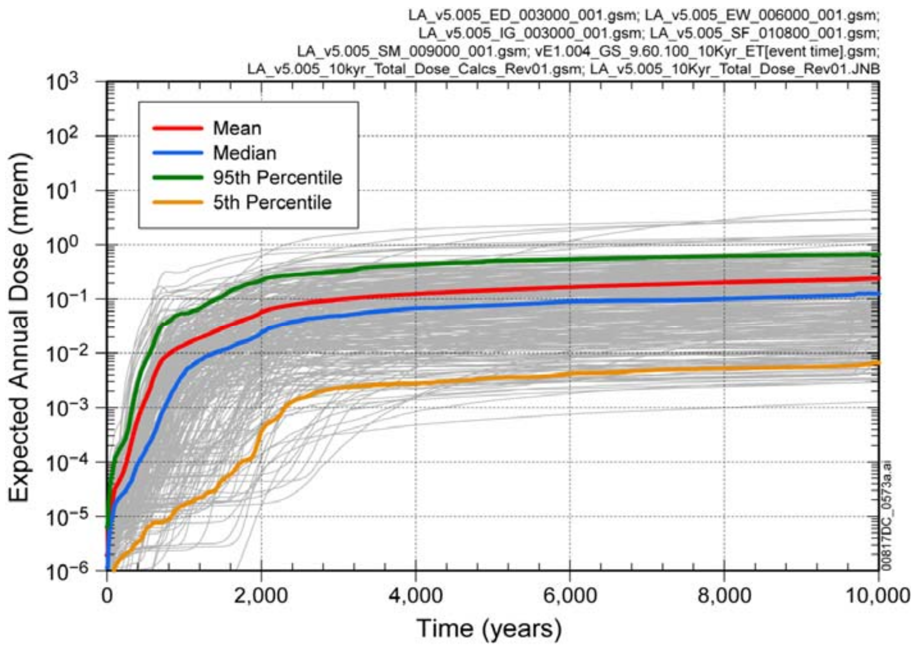

Figure 3. Yucca Mountain TSPA-LA 10,000-Year Results [20]

Greater Confinement Disposal (GCD) for DOE

Intermediate-depth boreholes (36 m deep), known as Greater Confinement Disposal (GCD) boreholes, were used to dispose of high-specific-activity LLW, sealed radioactive sources and limited quantities of classified TRU wastes in the Area 5 Radioactive Waste Management Site on the NTS in the 1980s. SNL evaluated the long-term performance of the GCD boreholes through 3 iterations of the GCD PA, which were conducted in 1993, 1994, and 2001 [25,26,27], to assess compliance with the containment requirements in 40 CFR Part 191. The final GCD PA [27] is significant in that it is the only approved safety assessment for intermediate-depth disposal of radioactive waste.

The GCD PA was conducted in an iterative fashion beginning with simple but defensible models. Four significant processes and events were identified including climate change, subsidence of the waste and overlying alluvial fill, and two scenarios involving inadvertent human intrusion. A total of 5,000 realizations of sampled probabilistic parameters were completed, and the resulting CCDF was well within the limits specified in 40 CFR 191. The GCD PA for these special case wastes was significant in that it was the first successful completion and acceptance of a PA for TRU waste under DOE self- regulation, and it was only the second site, after WIPP, to meet the safety requirements of 40 CFR 191 for disposal of TRU waste.

SNL also conducted a preliminary, or scoping, PA to evaluate the GCD boreholes at NTS for the disposal of vitrified Fernald byproduct material [28]. The byproduct material is concentrated residue from processing uranium ore, which has been stored at the Fernald Environmental Management Project since the early 1950s and will be vitrified prior to disposal. This preliminary PA identified issues and activities required to complete a full GCD PA for this waste stream.

APPLICATION OF PA METHODOLOGY TO THE BACK-END OF THE NUCLEAR FUEL CYCLE

The U.S. is currently faced with a massive nuclear waste disposal problem now that Yucca Mountain no longer an option for the disposal. Hence, a consistent, defensible, logical, and comprehensive approach is needed to manage the analyses and evaluations of the different strategies and options that might be proposed to address the waste problem. SNL's PA methodology provides a tool to carry out a total systems analysis to address this challenge, as well as a method for propagating the effect of important sources of uncertainties in the analysis. Sensitivity analyses would identify the most critical system components with respect to the performance measures. The result would be the identification of technically sound nuclear waste management options that reduce overall cost and prioritize funding activities by focusing efforts on what is most important to the performance measures. Recently, two preliminary PAs were conducted, one for deep borehole disposal [31] of HLW and SNF and one for the disposal of HLW in a generic salt repository [32]. Both PAs demonstrated technical feasibility worthy of further consideration. However, implementation of any alternative disposal strategy will require modifications to the Nuclear Waste Policy Act and could require modifications to 40 CFR Part 191 and 10 CFR Part 60.

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