PRECLOSURE SAFETY ANALYSIS: PHASE II EXERCISE

Prepared for

U.S. Nuclear Regulatory Commission
Contract NRC–02–02–012

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September 2007

ABSTRACT

The objective of this exercise was to develop risk insights for spent fuel pool operations at a conceptual Wet Handling Facility. Because of the potential risk significance of handling bare fuel, this exercise focused on handling operations in a conceptual spent fuel pool. Therefore, the operations associated with spent fuel pools were examined so that preclosure safety analysis team members could gain risk insights that could help them focus their efforts during the review of a potential license application for a geologic repository at Yucca Mountain, Nevada. This exercise was limited in scope to the analysis of hazards from internal operations that may take place at a conceptual Wet Handling Facility. Information on existing spent fuel pool operations and associated design was gathered to conceptualize a spent fuel pool for the potential Geologic Repository Operations Area at Yucca Mountain, Nevada. From this conceptualized model, potential hazards associated with lifting fuel assemblies were identified and an event sequence analysis with a corresponding consequence analysis was developed for a cask drop scenario.

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EXECUTIVE SUMMARY

The U.S. Department of Energy (DOE) is currently preparing to submit a License Application (LA) for a potential geological repository at Yucca Mountain, Nevada, to the U.S. Nuclear Regulatory Commission (NRC). In the prelicensing period, the NRC and the Center for Nuclear Waste Regulatory Analyses (CNWRA) staffs have been engaged in activities related to preclosure safety analysis for the Geological Repository Operations Area (GROA), in support of the future review of the DOE LA. At the potential repository, it is expected that spent nuclear fuel and high-level waste will be handled at surface facilities as part of operations leading to their permanent disposal in underground emplacement drifts.

In October 2005, DOE announced a major design change to the facilities at the Geological Repository Operations Area. Under the new design, the majority of the spent nuclear fuel would be loaded into transportation, aging, and disposal canisters at utility sites and shipped to the DOE Yucca Mountain Geological Repository Operations Area. The new design of the surface facilities also includes limited bare fuel handling in a spent fuel pool for fuel that cannot be readily packaged into the transportation, aging, and disposal canisters at the utility sites and, therefore, would need to be handled at the Geological Repository Operations Area. Because of the potential risk significance from handling bare fuel, the Preclosure Safety Analysis Phase II Exercise focused on handling operations in the spent fuel pool at a conceptual Wet Handling Facility.

The original scope of the Preclosure Safety Analysis Phase II Exercise involved (i) collecting information and conceptualizing a spent fuel pool facility; (ii) describing major operations in this conceptual facility; (iii) conducting an operational hazards analysis to identify hazards that could potentially lead to radiological consequences to workers and the public; (iv) conducting an event sequence analysis, including estimating reliability of structures, systems, and components relied on to prevent or mitigate the event sequence and categorizing the event sequences; and (v) performing consequence analysis for the event sequences developed in accordance to their categorization. It is emphasized that the analyses were performed only for a conceptual facility.

Under NRC guidance, the Preclosure Safety Analysis Phase II Exercise scope of work was revised to document work that had been completed or is near completion. This report describes progress achieved to date on the Preclosure Safety Analysis Phase II Exercise. This includes information collection, conceptualization of a conceptual Wet Handling Facility, operational hazard analysis, event sequence analysis, and dose consequence analysis for a conceptual Wet Handling Facility involving the drop of a shielded transfer cask containing a loaded transportation, aging, and disposal canister due to crane failure as an initiating event.

The initiating event frequency analysis estimated the unmitigated drop of a transfer cask to be a Category 2 event sequence with an estimated 0.167 drops over the 30-year operational period. Uncertainty analysis of the initiating event frequency, however, revealed that at the 95 percent confidence interval upper limit, the estimated expected number of drops was 0.5. If a sensitivity analysis is included, then the expected number of drops was estimated to be 0.95 at the upper limit of the 95 percent confidence interval, and the expected number of drops was estimated to be 1.11 over the assumed 30-year operational period at the 99 percent confidence interval upper limit. Heating, ventilation, and air-conditioning exhaust system reliability was considered to develop the overall event sequence frequencies.

A radiological consequence analysis was performed for workers in the vicinity of the cask drop and workers located outside the building at a distance of 200 m [656 ft]. No consequence analysis was performed for the public. The analysis of potential radiological consequences does not include the probability of the cask drop event, the potential drop height, or the fraction of the fuel rods damaged as a result of the drop; therefore, the analysis should be considered for illustration purposes only.

ACKNOWLEDGMENTS

This report was prepared to document work performed by the Center for Nuclear Waste Regulatory Analyses (CNWRA) and the U.S. Nuclear Regulatory Commission (NRC) under Contract No. NRC–02–02–012. The activities reported here were performed on behalf of the NRC Office of Nuclear Material Safety and Safeguards, Division of High-Level Waste Repository Safety. This report is a joint product of the CNWRA and NRC and does not necessarily reflect the view or regulatory position of the NRC. The NRC staff views expressed herein are preliminary and do not constitute a final judgment or determination of the matters addressed or of the acceptability of a license application for a geologic repository at Yucca Mountain.

The authors thank B. Dasgupta for his assistance in reviewing Chapter 3, T. Ghosh and S. Cooper for their insights into human reliability analysis, and S. Whaley and R. Johnson for their technical and programmatic inputs. We also thank A. Ghosh, S. Hsiung, and J. Durham for their technical reviews and W. Patrick for his programmatic review. Thanks are also expressed to L. Mulverhill for her editorial review and R. Mantooth and A. Ramos for secretarial support.

QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

DATA: All CNWRA-generated original data contained in this report meet the quality assurance requirements described in the Quality Assurance Manual. The modeling work is documented in CNWRA scientific notebooks number 658 and 875E. Data used in this report are primarily derived from other publicly available sources. Each data source is cited in this report and should be consulted to determine the level of quality for those cited data.

ANALYSES AND CODES: Dose consequence analyses in this report were conducted by CNWRA using the PCSA Tool Version 3.0.1 BetaC (Maxwell et al., 2005), which is controlled under the software quality assurance procedure Technical Operating Procedure—018, Development and Control of Scientific and Engineering Software (GED, 2005). Event Sequence Analysis was conducted using MATLAB Version 7.1.0.246 (R14) Service Pack 3 (Mathworks, 2005) and Mathcad 2000 Professional (MathSoft, 1999).

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1 INTRODUCTION

1.1 Background

The U.S. Department of Energy (DOE) is currently preparing to submit a License Application for a potential geological repository at Yucca Mountain, Nevada, to the U.S. Nuclear Regulatory Commission (NRC). In the prelicensing period, the NRC and the Center for Nuclear Waste Regulatory Analyses (CNWRA) staffs have been engaged in activities related to preclosure safety analysis for the Geological Repository Operations Area, in support of the future review of the DOE License Application. At the potential repository, it is expected that spent nuclear fuel and high-level waste will be handled at surface facilities as part of the operations leading to their permanent disposal in underground emplacement drifts.

Between 2005 and 2006, NRC and CNWRA staffs conducted a limited preclosure safety analysis for a hypothetical fuel handling facility. DOE had previously envisioned dry fuel handling operations in a fuel handling facility for the potential Geologic Repository Operations Area. During this Preclosure Safety Analysis Phase I Exercise, a hypothetical fuel handling facility was analyzed and staff identified potential event sequences related to handling canisters and bare fuel assemblies. The Preclosure Safety Analysis Phase I Exercise was not documented as a report.

In October 2005, DOE announced a major design change to the facilities at the Geological Repository Operations Area. Under the new design, the majority of the spent nuclear fuel would be loaded into transportation, aging, and disposal canisters at utility sites and shipped to the DOE Yucca Mountain Geological Repository Operations Area. The new design of the surface facilities also includes limited bare fuel handling in a spent fuel pool for fuel that cannot be readily packaged into the transportation, aging, and disposal canisters at the utility sites and, therefore, would need to be handled at the Geological Repository Operations Area. Because of the potential risk significance from handling bare fuel, the Preclosure Safety Analysis Phase II Exercise focused on handling operations in the spent fuel pool at a conceptual Wet Handling Facility. In addition, human reliability was identified in the Phase I Exercise as potentially risk-significant because of the intensive human interactions expected in handling the spent nuclear fuel. Therefore, human reliability was originally considered to be an important topic for the Preclosure Safety Analysis Phase II Exercise.

1.2 Objectives and Scope

The original scope of the exercise involved development of risk insights associated with bare fuel handling in a spent fuel pool, including human reliability analysis as an important component. Based on NRC direction received on June 11, 2007 (Johnson, 2007), the scope was revised to exclude human reliability analyses or risk insights. The revised report discusses the work that was completed by the team after the scope was revised. A more detailed discussion of the original scope and the revised scope follow.

1.2.1 Original Scope

The exercise was intended to be guided by the Preclosure Safety Analysis review guidelines outlined in Sections 2.1.1.2 through 2.1.1.6 of the Yucca Mountain Review Plan (NRC, 2003b)

and other suitable NRC guidance documents. Appropriate tools, such as the PCSA Tool software (Maxwell et al., 2005), would be used to conduct this exercise. Results of the exercise would then be documented in a report. Staff would collect currently available information to conceptualize major operations (e.g., cask lifting, ventilation) expected to take place at the facility with the transportation, aging, and disposal canister handling and spent nuclear fuel handling in a pool. As necessary, staff would make assumptions based on accepted engineering practice to fill in the gaps if actual design information directly applicable to the potential geologic repository facilities was not available. Staff would develop risk insights for the surface facility operations by including human reliability and operating experience from analogous facilities, because human operations were identified in the Phase I Exercise to be potentially risk-significant. Therefore, human reliability was conceived to be a central topic in the Phase II Exercise. The original scope included a division of the work in six major tasks.

- Task 1—Collect Information and Conceptualize the Spent Fuel Pool: Information would be gathered primarily from NUREGs and the August 29, 2006 NRC/DOE Technical Exchange and Management Meeting presentation slides (Tooker and Dunn, 2006) to understand the operations of a spent fuel pool and conceptualize such a facility.

- Task 2—Describe Major Operations (NRC, 2003b, Section 2.1.1.2): The major operations would be identified for the conceptual spent fuel pool, including lifting, transfering, loading, and closure operations.

- Task 3—Conduct Operational Hazards Analysis (NRC, 2003b, Section 2.1.1.3): A preliminary hazards analysis for the conceptual spent fuel pool would be conducted to identify hazards that could potentially lead to radiological consequences to workers or the public.

- Task 4—Conduct Event Sequence Analysis (NRC, 2003b, Section 2.1.1.4): Potential event sequences would be postulated and categorized based on estimated frequency of occurrence. The estimated event sequence frequency would be determined from initiating event frequencies and reliability estimates of structures, systems, and components relied on to prevent or mitigate the event sequence.

- Task 5—Perform Consequence Analysis (NRC, 2003b, Section 2.1.1.5): Radiological dose would be calculated to the offsite public for postulated normal operations, and postulated Category 1 and Category 2 event sequences. Radiological dose would be calculated to workers for postulated category 1 event sequences using the PCSA Tool.

- Task 6—Document the Results and Risk Insights: The results and risk insights would be documented as part of a report.

1.2.2 Revised Scope

Based on NRC direction documented in the Preclosure Safety Analysis Phase II Exercise Meeting Summary, June 11, 2007, which was included as an attachment to an e-mail (Johnson, 2007), the scope of the Phase II Exercise was modified as follows.

- The PCSA Exercise deliverable was refocused to document work that was completed or near completion. This report would be a CNWRA intermediate milestone deliverable to

NRC. Minimal effort was spent on the Phase II Exercise to complete the documentation. The team assembled sections that had been completed or were near completion, and developed a report from those sections. Any section (e.g., Human Reliability Analysis and risk insights) that had not been developed or would require a substantial effort to complete was removed from the report.

- In addition, the revised CNWRA deliverable was conceived as a letter describing the scope of the effort and a report documenting the valuable work that was completed (or was near completion).

The work breakdown structure for this scope and the associated report are summarized as follows.

- Task 1—Review currently available information and conceptualize a wet handling facility. This information is included in Chapter 2, Section 2.1.

- Task 2—Describe major operations and basic functions within the conceptualized facility. The description of major operations was reduced in scope to lifting operations only and is included in Chapter 2, Section 2.2.

- Task 3—Conduct operational hazard analysis. The operational hazard analysis is included in Chapter 3.

- Task 4—Conduct event sequence analysis. The event sequence analysis is included in Chapter 4. Based on NRC direction, only a single initiating event was considered. This scenario involved dropping a cask that was being transferred from the spent fuel pool to the welding area. This analysis is included in Chapter 4.

- Task 5—Perform consequence analysis. A consequence analysis was performed in Chapter 5 for the single initiating event considered in Chapter 4.

- Task 6—Document the intermediate results on an ongoing basis. As a result of the revised NRC direction, only the intermediate results were documented. No broader risk insights were developed.

2 METHODOLOGY FOR PRECLOSURE SAFETY ANALYSIS

2.1 Information Collection and Conceptualization (Task 1)

Information on spent fuel operations primarily was collected from NUREGs and standards. In addition, a regulatory guide in which spent fuel pool operations are described was consulted. Presentation slides of the August 29, 2006, NRC/DOE Technical Exchange and Management Meeting on design changes approved through the DOE Critical Decision-1 Process (Tooker and Dunn, 2006), in which DOE described the conceptual Wet Handling Facility for the potential Geologic Repository Operations Area, were also considered. Information from published literature was used because the spent fuel pool operations at the potential geological repository are anticipated to be similar to operations at commercial nuclear power plants. Section 2.1.1 describes the information sources, and Section 2.1.2 describes the conceptual facility.

2.1.1 Information Collection

Information collected from several data sources is included in this section. Many of the data sources identified the potential drop of a heavy load onto the spent fuel pool structure as an initiating event. Other events were also identified for spent fuel pool operations and are included so that they could be considered in subsequent tasks. The data sources generally presented information for spent fuel pools associated with reactor sites. Therefore, the spent fuel pools described in these data sources considered fuel that had been out of a reactor core for a much shorter time than would be the case for spent fuel at the conceptual spent fuel pool for this exercise. For this reason, some of the events identified in these data sources would not be applicable to this conceptual spent fuel pool, but the information is included so that it can be considered in future analyses.

2.1.1.1 NUREG–1275, Volume 12, Operating Experience Feedback Report—Assessment for Spent Fuel Cooling (NRC, 1997a)

The major scenarios associated with spent fuel pools in NUREG–1275 involve loss of coolant inventory from the spent fuel pool and loss of cooling to the spent fuel pool. Additional scenarios were identified in which flooding occurred in other areas of the plant from spent fuel pool overflow as a result of human error. In some cases, the spent fuel pool overflowed into the ventilation system or the reactor building. Flooding was identified as having the potential to affect other plant equipment.

For the loss-of-coolant inventory scenario, the identified causes were the drop of a heavy load resulting in gross failure of the spent fuel pool structure and loss of coolant due to siphoning. Only one actual drop event occurred in which the spent fuel pool liner was punctured; however, more than 30 precursor load drop events were identified in which damage could have occurred to the spent fuel pool. Siphoning occurred as a result of plugging of a nonredundant vent line and use of temporary equipment (i.e., a fill pump). In addition, a precursor event was observed in which antisiphon holes thought to be present in cooling return lines were not actually present.

The primary cause for loss of spent fuel pool cooling was the loss of electrical power to the spent fuel pool cooling pumps. Human errors and administrative problems were identified as dominant contributors to this failure mode. In addition, when spent fuel pool cooling was lost for extended periods, vapor could be transported to other locations and condense, thereby affecting other equipment.

2.1.1.2 NUREG/CR–4982, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82 (NRC, 1987)

This NUREG identifies four major contributors to the estimated accident frequency in a spent fuel pool: (i) cask drop accidents, (ii) seismic-induced pool failure, (iii) loss of pool cooling, and (iv) pneumatic seal failure. Because seismic events are not considered in this exercise and use of pneumatic seals is not applicable to the spent fuel pool in the conceptual facility, only cask drop accidents and the loss of pool cooling are considered further.

This document indicates that to eliminate the risk of a cask dropping and striking the edge of the main pool, some spent fuel pools have a special section for the cask, which is separated from the main pool by a weir or gate. For such a design, the number of times a cask is transferred over the edge of the main pool would be zero, thereby eliminating the potential for a dropped cask to damage the main pool.

2.1.1.3 NUREG–1353, Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools (NRC, 1989)

This NUREG reexamines spent fuel storage pool accidents stemming from the use of high density storage racks and consideration of laboratory studies indicating the possibility for zircaloy cladding fire propagation between assemblies in an air-cooled environment. This NUREG indicates that “for a zircaloy cladding fire to occur, the fuel must be recently discharged (between 30 and 180 days in a cylindrical boiling water reactor configuration, and between 30 and 250 days in a cylindrical pressurized water reactor configuration)” (NRC, 1989). High density storage racks are assumed to not be used in the conceptual spent fuel pool. In addition, the spent fuel that would be received in this conceptual spent fuel pool was assumed to be at least 5 years old; therefore, cladding fires were not considered further in this exercise.

This NUREG provides useful insights such as the inadvertent draining of the spent fuel pool. It indicates that pipe breaks in the cooling system, heat exchangers, or siphoning paths could result in loss of water. It indicates that a number of incidents have occurred that have been identified in the Information Notice 88-65, “Inadvertent Drainages of Spent Fuel Pools” (NRC, 1988). One incident in particular involves the San Onofre 2 site (June 22, 1988) in which a siphoning path was present in the purification system resulting in 34,069 liters [9,000 gal] siphoned. It indicates that although siphon breakers, check valves, and locked valves were installed, administrative controls were not established to prevent alignment of the cooling system; consequently, the siphoning event occurred. However, the NUREG indicates that this event did not result in a significant loss of water from the spent fuel pool, because the 34,069 liters [9,000 gal] were equivalent to only about a 50-cm [19.5-in] drop in the pool water level.

In addition, a series of spent fuel pool review guidelines and requirements with regard to the spent fuel pool safety functions were identified. These include (i) cooling spent fuel assemblies, (ii) maintaining them in a subcritical array during storage, and (iii) providing a safe means of loading assemblies into shipping casks. The NUREG indicates that a safety review for such a facility would include, among other things, the following considerations:

- Ability to maintain pool water level
- "Provisions to provide adequate make-up to the pool" (NRC, 1989)
- "Provisions to preclude loss of function resulting from single active failures or failures of non-safety related components or systems" (NRC, 1989)
- "The means provided for the detection and isolation of system components that could develop leaks or failures" (NRC, 1989)

2.1.1.4 NUREG/CR–5176, Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants (LLNL, 1989)

This document addresses seismic failure of the spent fuel pool, which is not considered in this exercise. However, this NUREG does address a cask drop on the wall of a spent fuel pool in one facility that has a boiling water reactor spent fuel pool and another facility that has a pressurized water reactor spent fuel pool. For the cask drop analysis, it was assumed that, in the event of a cask drop, "…the conditional probability of structure failure of the spent fuel pool wall given a cask drop has a value of 0.1 with an uncertainty range of 0.01 to 1" (LLNL, 1989). This NUREG also references Generic Letter 85-11 (NRC, 1985), which requires licensees to meet specific criteria "…such that the risk associated with potential heavy load drops is acceptably small" (NRC, 1985) considering "(i) single failure cranes, (ii) appropriate handling procedures and training, and (iii) analysis of heavy load structural and mechanical capacities" (LLNL, 1989).

For the pressurized water reactor spent fuel pool, an analysis was performed for a 61.7 tonne [68 ton] cask dropping from a height of 1.2 m [4 ft]. The peripheral walls were 1.8-m [6-ft]-thick reinforced concrete. Results of this analysis indicated that (i) the pool walls would suffer severe damage, and (ii) "…while the integrity of the pool liner is difficult to predict, it seems likely that the liner would be severely damaged" (LLNL, 1989).

2.1.1.5 NUREG–1738, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (NRC, 2001)

This report documents a study of risk associated with spent fuel pool accidents at decommissioning nuclear power plants. A modeling approach based on developing qualitative and quantitative assessments from available information is described for a typical decommissioning plant, bounded by design assumptions and industry commitments. Due to the general nature of the study, which was prepared to support rulemaking affecting the decommissioning activities of nuclear power plants, application of the findings to spent fuel pool facilities depends on design and operational characteristics assumed in the analysis. These considerations are similar to the existing constraints for the Preclosure Safety Analysis Exercise Phase II, where general assumptions are used instead of specific design details. However, this study was performed for a different application (i.e., spent fuel pools at nuclear power plants)

than what is expected for the Geologic Repository Operations Area and, therefore, its applicability would need to be carefully assessed.

In this document, several types of analyses were performed that included thermal-hydraulic analyses, risk assessment, consequence calculation, human reliability analysis, and sensitivity studies based on probabilistic risk assessment methods. To develop risk estimates, a number of assumptions were made and documented as industry decommissioning commitments and staff decommissioning assumptions. Industry decommissioning commitments were derived to reflect actual operating practices at decommissioning facilities based on discussions with the nuclear industry via the Nuclear Energy Institute. Staff decommissioning assumptions were necessary to support the derived probability values for certain scenarios, where industry decommissioning commitments were not sufficient to achieve this objective. The following initiating events were identified for risk assessment analyses in NUREG–1738:

- Heavy load drop
- Loss of coolant inventory
- Loss of cooling for the pool
- Criticality
- Seismic event
- Aircraft crash
- Tornado missile
- Loss of offsite electrical power due to plant- and grid-related events
- Loss of offsite electrical power due to severe weather
- Internal fire

This study concluded that the largest contributors to risk in spent fuel pools at decommissioning nuclear power plants are the event sequences related to large seismic events and heavy load drop scenarios. Most severe accidents are related to loss of coolant in the pool (i.e., loss of water). As external hazards and internal fire were not considered in this exercise, only the remaining initiators were investigated for potential input information into the Preclosure Safety Analysis Phase II Exercise.

For loss of cooling as an initiating event due to mechanical failure in the cooling system or due to loss of electrical power, the analysis presented in NUREG–1738 (NRC, 2001) indicated that the initiating frequency is very low and there is a long time available for recovery. In this case, the considered scenario assumes that loss of cooling leads to a heat increase (i.e., heat up) in the pool and eventual boiling of the remaining coolant (i.e., boil off) by the fuel decay heat. The critical parameter in this analysis was established to be the time needed to lower the pool level down to 0.9 m [3 ft] above the top of the fuel stored in the pool. For 5-year-old fuel, it was calculated that at least 16 days would be needed for heatup and boiloff to occur for pressurized water reactor fuel and 19 days for boiling water reactor fuel.

The loss-of-coolant scenario discussion in NUREG–1738 (NRC, 2001) refers to the assumptions presented in NUREG–1275 (NRC, 1997a, Vol. 12) (see Section 2.1.1.1). The frequency of fuel uncovery in this scenario is attributed to configuration control errors and failure to contain the coolant (probable causes could be, for example, piping or seal failure) and is assumed to be low on the basis that loss of coolant alone is not expected to fully drain the pool (unless structural damage is also considered). Again, the time for recovery is estimated to be approximately 40 hours, even if a significant leak occurs. This study identifies two recent

events (one in 1998 and another in 2000) where a short-duration loss-of-cooling event leading to limited temperature increase occurred in spent fuel pools at two different nuclear power plants.

The load drop scenarios in the spent fuel pool, as reported in this document, include a heavy load drop into the pool or onto the pool wall due to a crane failure (mechanical or human related) leading to the unexpected release of the load. The results from this initiating event were considered in terms of the conditional probability of structural damage in the event of a drop, resulting in a subsequent loss of coolant inventory. The analysis used in NUREG–1738 (NRC, 2001) takes into account considerations described in NUREG–0612 (NRC, 1980). Since the publication of NUREG–1738, NUREG–1774 (NRC, 2003a) was released, which is considered both a more recent and a more rigorous investigation of heavy load drops due to crane failure.

For the risk analysis performed in NUREG–1738, point estimates (i.e., assumed as equivalent to the mean of the unquantified probability distributions) were deemed to be sufficient for input in the risk analysis with no further uncertainty propagation because of the large margin between event sequence frequencies and the performance limits established for the purposes of the study.

2.1.1.6 ANSI/ANS–57.7–1988 (Reaffirmed 1997), Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type) (ANS, 1988)

This standard provides some insights into (i) design features for cranes that may be used in a spent fuel pool, (ii) design features for equipment that may be used in a spent fuel pool building, and (iii) the layout of a spent fuel pool building.

For cranes that may be used at a spent fuel pool, Section 6.5 addresses fuel unit (i.e., spent fuel assembly, canned spent fuel, or consolidated fuel rods) handling and indicates in Section 6.5.2.6 that “Fuel handling equipment used for fuel unit unloading shall have an overload interlock actuated upon an unacceptable increase in hoisting force to prevent any further upward travel.” (ANS, 1988) This requirement may be important in preventing fuel assembly damage, for example, in event sequences where a fuel assembly is stuck in a cask or in a storage rack. It indicates in Section 6.5.2.17 that “Vertical travel of the fuel handling equipment shall be positively restricted to assure maintenance of the minimum watershielding over the fuel assembly.” (ANS, 1988) If the crane meets this requirement, then workers in the vicinity of the spent fuel pool may be protected from exceeding their dose limits from an assembly pulled too close to the surface of the water. This requirement may have prevented the event documented in NUREG-1774 in which “the crane lifted [an] irradiated fuel assembly out of the spent fuel pool resulting in increased exposure.” (NRC, 2003a)

For heating, ventilation, and air conditioning associated with a spent fuel pool, Section 6.6 indicates that the heating, ventilation, and air-conditioning system “…shall consist of subsystems based on the level of potential for airborne radioactive material contamination.” (ANS, 1988) With regard to flow of air in a spent fuel pool building, Section 6.6.2.1.2 states that “The subsystems shall be designed to provide for flow of air from areas of low potential for radioactive contamination to areas of higher potential for radioactive contamination.” (ANS, 1988) These heating, ventilation, and air conditioning requirements indicate that the spent fuel pool building would be separated into areas based on the level of potential for airborne

radioactive material contamination and that maintaining flow from areas of low potential for radioactive material contamination to areas of higher potential is important. Therefore the heating, ventilation, and air-conditioning system is anticipated to have a function to maintain differential pressure between these areas and reviewers of a license application may need to consider the reliability of the heating, ventilation, and air-conditioning system for maintaining differential pressure for event sequences involving cask or canister drops in the vicinity of the pool.

This standard also provides typical general arrangement diagrams for pools showing a layout in which the fuel pool is separated from the cask unloading area (i.e., cask pool) by a swinging gate. This layout is similar to what was described in Section 2.1.1.2 for which the potential of a cask drop to damage the main pool was eliminated because the cask would not be transferred over the edge of the main pool. These diagrams also indicate that the cask pool contains a shelf structure on one side. The presence of a shelf in the spent fuel pool was also observed on the DOE presentation slides (Tooker and Dunn, 2006).

2.1.1.7 ANSI/ANS–57.1–1992 (Reaffirmed 2005), Design Requirements for Light Water Reactor Fuel Handling Systems (ANS, 1992)

This standard establishes the required functions of fuel handling systems at light water reactor nuclear power plants. "It provides minimum design requirements for equipment and tools to handle nuclear fuel and control components [e.g., control rods, burnable poison rods] safely" (ANS, 1992). Section 6.3.1.1 specifies the required interlock protection for a fuel handling machine (i.e., a machine that operates over the spent fuel pool and handles fuel units and control components) and other equipment. It indicates that these "features [i.e., interlocks] shall be provided…to prevent damage to fuel units or control components and to provide for personnel safety" (ANS, 1992). However, this standard allows manual bypasses for interlocks to be supplied at the discretion of the designer. The importance of this provision is that if an interlock is inadvertently left bypassed, for example due to human error, then the interlock may not be available to perform its safety function when required. Reviewers of a license application may need to consider what controls are in place to ensure that an interlock is not inadvertently left bypassed. Such an event occurred where an operator failed to reinstate a hoist height interlock prior to moving a load over the spent fuel pool racks containing fuel (Pollock, 2002). In this case, "…the procedure did not contain specific steps for reinstating the interlocks" (Pollock, 2002).

2.1.2 Development of Conceptual Wet Handling Facility

The conceptual Wet Handling Facility for a potential repository was conceptualized using publicly available information. Because detailed design information for proposed surface facilities was not available from DOE at the time of this PCSA Phase II Exercise, the primary source of information came from the presentations and discussions at the August 29, 2006, NRC/DOE Technical Exchange and Management Meeting on design changes approved through the DOE Critical Decision-1 Process (Tooker and Dunn, 2006). This conceptualization was complemented, where applicable, with information available in some of the references described in Section 2.1.1.

2.1.2.1 Facility Layout

Figure 2-1 shows the conceptual facility layout used in this PCSA Phase II Exercise for the conceptual Wet Handling Facility. This facility is conceptualized to include the spent fuel pool and structures, systems, and components related to handling of the casks and spent fuel assemblies, along with any other supporting operations. The larger main room of the facility is where the spent fuel pool is located, along with areas for specific handling activities, described as preparation area, handling area, and welding area. Operations expected to be performed in these areas include gas sampling, bolting and unbolting of cask lids, cutting and welding of canisters, and cask lifting. Underwater handling and temporary storage of bare spent fuel assemblies take place in the spent fuel pool.

Receiving and shipping operations occur through either the main vestibule or the side vestibule. The main vestibule receives spent fuel in transportation casks from either rail or truck transportation, while the side vestibule can be used to transport canisters to and from the aging facility, whose conceptualization was beyond the scope of the Phase II Exercise.

Other structures, systems, and components assumed to be located in areas adjacent to the main room include the heating, ventilation, and air conditioning system and the supporting electrical power supply system. A control room for monitoring activities in the overall facility is also indicated in Figure 2-1, along with a pool equipment area that can be used for storing components (e.g., filters, pumps) related to the pool cooling system.

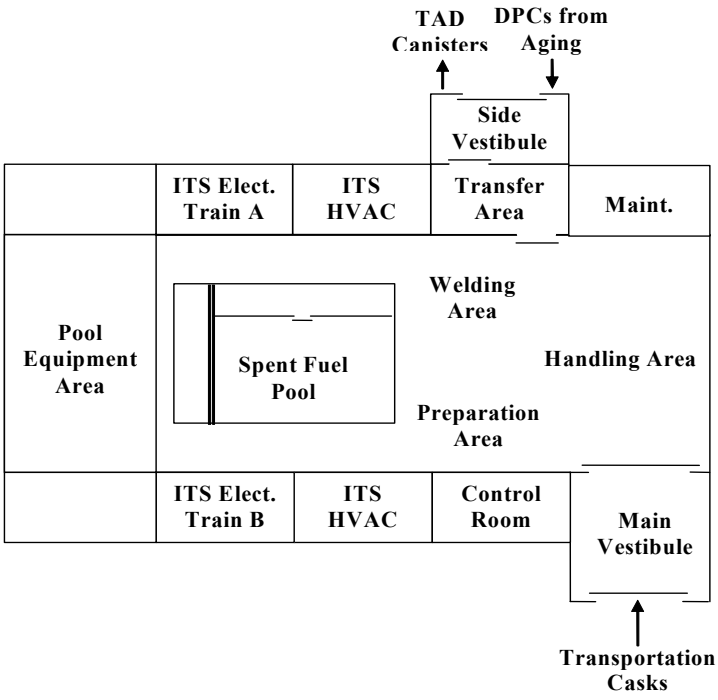


Figure 2-1. Conceptual Wet Handling Facility (ITS: Important to Safety; HVAC: Heating, Ventilation, and Air Conditioning; TAD: Transportation, Aging, and Disposal; DPC: Dual Purpose Canister)

The spent fuel pool is assumed to be 19 m [61 ft] wide by 23 m [75 ft] long, with a maximum depth of 15 m [50 ft], given the dimensions provided in Tooker and Dunn (2006). Based on these dimensions, the major overall dimensions for the conceptual Wet Handling Facility are extrapolated to be 52-m [170 ft] wide by 82 m [270 ft] long, not including the extended main vestibule and side vestibule areas. Similarly, the height measured from the pool to the ceiling of the main room area is extrapolated as 24.4 m [80 ft].

2.1.2.2 Lifting Equipment

The conceptual Wet Handling Facility uses three separate Type I cranes designed according to the American Society of Mechanical Engineers NOG-1 (ASME, 2005): (i) cask handling crane, (ii) canister transfer machine, and (iii) spent fuel transfer machine. An overhead bridge crane is used to lift casks and transfer them to and from the spent fuel pool, welding area, and preparation area. The canister transfer machine is used to lift a transportation, aging, and disposal canister contained in a shielded transfer cask on the cask trolley and transfer it to an aging overpack. It is also used to lift a dual purpose canister contained in an aging overpack and transfer the dual purpose canister into a shielded transfer cask. The spent fuel transfer machine is used to transfer fuel assemblies in the spent fuel pool to a transportation, aging, and disposal canister and to and from the storage racks. It is also used to transfer fuel assemblies from a dual purpose canister that has been cut open to either a transportation, aging, and disposal canister or to the storage racks. A fourth crane lifts transportation casks in the handling area. However, this fourth crane and the canister transfer machine are not considered further in this exercise because the focus is on operations that are particular to the spent fuel pool (e.g., handling of bare fuel assemblies, lifting canisters from the pool).

2.1.2.3 Spent Fuel Pool

The conceptual spent fuel pool is an unborated fuel pool with storage racks for individual fuel assemblies and a separate unloading area for dual purpose canisters. The following discussion is based on a conceptual design for the spent fuel pool that was developed in this exercise. It has been adapted from the August 29, 2006, NRC/DOE Technical Exchange and Management Meeting presentation slides (Tooker and Dunn, 2006).

For the conceptual spent fuel pool design shown in Figure 2-2, a gate is used to separate the transportation cask unloading area and transportation, aging, and disposal loading area from the adjacent dual purpose canister unloading area. Fuel is transferred under water from a transportation cask or dual purpose canister to either a storage rack or to a transportation, aging, and disposal canister using a spent fuel transfer machine. The gate, located in a partition wall, is designed to contain contaminated water from the dual purpose canister to one area of the spent fuel pool.

For the design shown in Figure 2-2, the cask handling crane is used to transfer loaded transportation casks or dual purpose canisters to their unloading areas. The same bridge crane is used to transfer a shielded transfer cask containing an empty transportation, aging, and disposal canister into the transportation, aging, and disposal loading area and remove the loaded transfer cask from this area. The staging shelf is the last position occupied by the transportation, aging, and disposal canister that has been placed in a shielded transfer cask

before it is removed from the pool; it is the first position of the transportation cask or dual purpose canister/shielded transfer cask when it is placed in the pool. A short lifting yoke is used on the cask handling crane to place a cask onto the staging shelf. A long lifting yoke is used to position the cask into a loading or unloading area of the pool. The use of different lifting yokes allows the crane hoisting mechanisms to remain out of the pool water.

2.1.2.4 Transportation, Aging, and Disposal Canister System

The current DOE design approach for transportation, handling, storage, and disposal of commercial spent nuclear fuel is to use a canister-based system for the potential Geologic Repository Operations Area. This approach includes the transportation, aging, and disposal canister; the shielded transfer cask; and various overpacks to be used in different functions (transportation, aging, and disposal). General information available from DOE (Zabransky, 2006) is summarized next, emphasizing the parameters that are most relevant to the work performed in this exercise. Table 2-1 summarizes the values for those parameters, as defined by DOE (Zabransky, 2006).

The transportation, aging, and disposal canister system is anticipated to be comprised of a canister shell, lids, and internal components, such as a basket for holding spent nuclear fuel assemblies, thermal shunts, and fixed neutron absorbers. This canister is loaded with commercial spent nuclear fuel and sealed either at the Geologic Repository Operations Area or at facilities outside the potential repository (e.g., nuclear power plants). If loaded other than at the potential repository, the transportation, aging, and disposal canister would be shipped to the Geologic Repository Operations Area in a transportation overpack. According to Zabransky

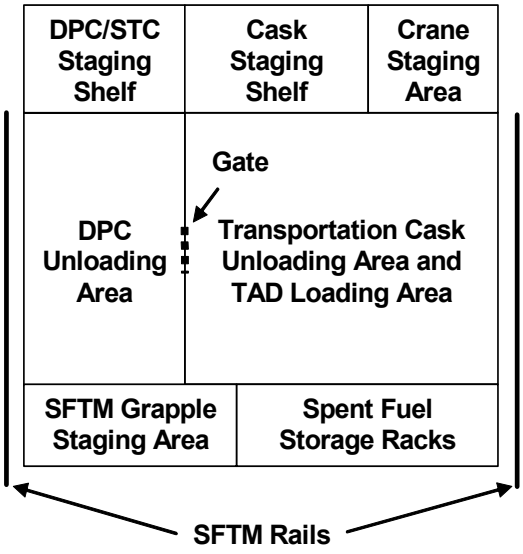


Figure 2-2. Conceptual Spent Fuel Pool Layout (DPC: Dual Purpose Canister; STC: Shielded Transfer Cask; SFTM: Spent Fuel Transfer Machine; TAD: Transportation, Aging, and Disposal)

(2006), at the Geologic Repository Operations Area, a transportation, aging, and disposal canister "may also be handled using a shielded transfer cask or aged in an aging overpack; and shall be disposed of in a waste package." Furthermore, DOE indicates in this same reference that the loaded and closed transportation, aging, and disposal canister is anticipated to be designed for certain operations in the pool, such as reopening while submerged.

DOE defines the burnup limits for commercial spent nuclear fuel as 80 Gwd/MTU or less for pressurized water reactors and 75 GWd/MTU or less for boiling water reactors. In both cases, the commercial spent nuclear fuel characteristics are defined as less than 5 percent initial enrichment burnup and a minimum of 5 years out-of-reactor cooling time. The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME, 2004) is identified by DOE (Zabransky, 2006) as the standard for design, procedure, and qualification requirements of the transportation, aging, and disposal canister. The closure welds for the transportation, aging, and disposal canister will be in accordance with standard nuclear industry practice.

The transportation overpack will be certified under 10 CFR Part 71 as a packaging component used to enclose transportation, aging, and disposal canisters for transportation from outside facilities to the Geologic Repository Operations Area. The anticipated functions accredited to the transportation overpack in Zabransky (2006) are protection during normal conditions of transport and design basis accidents, decay heat dissipation from the commercial spent nuclear fuel, and radiation protection. The transportation overpack is anticipated to have trunnions that permit lifting by an overhead crane, so it can be upended during handling operations. Other lifting features could be present to allow for the same common devices to be used between a

Table 2-1. Parameters for the Transportation, Aging, and Disposal Canister System*

	Length (cm) [inches]	**Diameter (cm) [inches]**	**Max. Loaded Weight (tonne) [tons]**	**Capacity**	**Minimum Service Lifetime**
Transportation, Aging, and Disposal Canister	538.5 (+0,−1.3) [212.0 (+0,−0.5)]	168.9 (+0,−1.3) [66.5 (+0,−0.5)]	58.3 [54.25]	21 PWR/ 44 BWR assemblies	100 years†
Transportation Overpack	584 or 846‡ [230 or 333]‡	249 or 320§ [98 or 126]§	227 [250]	Maximum Loaded TAD Canister	—
Aging Overpack	671 [264]	213 or 353¶ [84 or 139]¶	181 [200]	Maximum Loaded TAD Canister	100 years

*Zabransky, D.K. "Preliminary Transportation, Aging and Disposal Canister System Performance Specification, Revision B." Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2006.
†Prior to emplacement for disposal, without maintenance
‡Maximum values given without and with impact limiters, respectively
§Maximum values for cask lid diameter or impact limiters, respectively
¶Maximum values for overpack lid diameter or overpack diameter, respective

transportation overpack and a loaded transportation, aging, and disposal canister (e.g., grapple) (Zabransky, 2006).

An aging overpack will be used to temporarily contain a loaded transportation, aging, and disposal canister in the aging facility in a vertical orientation. The anticipated functions of the transportation overpack are protection from damage, decay heat dissipation, and radiation protection (Zabransky, 2006). The loaded aging overpack is also anticipated to be handled in a vertical orientation, to remain stable in this position when placed in a flat horizontal surface, and to be suitable for lifting operations by an overhead crane.

The shielded transfer cask will be used to transport a loaded transportation, aging, and disposal canister between different facilities at the Geologic Repository Operations Area (unless conditions for aging or emplacement are met, in which case an aging overpack or waste package will be used, respectively). The anticipated functions for the shielded transfer cask are protection of the transportation, aging, and disposal canister from damage, heat dissipation, and radiation protection.

2.1.2.5 Electrical System

From discussions with DOE during technical exchange meetings, the electrical system will be a "once through" system that is operated continuously; it does not change state when radioactive material is detected. From discussions with DOE (Tooker and Dunn, 2006), a plausible electrical system can be postulated, as illustrated in Figure 2-3. The system consists of (i) offsite power lines; (ii) diesel generator power system; (iii) main electrical bus; (iv) electrical loads; (v) battery charger; (vi) inverter; and (vii) instrument and control systems.

Power would normally be supplied from one of the offsite power lines, through the main electrical bus, to the electrical loads throughout the facility. Instrument and control systems would be powered through the battery charger that also keeps the batteries charged.

If power were to be lost from one offsite line, a sense and command system closes a switch to transfer the load on the other offsite line. If there is still no power on the main bus, the loads would be disconnected from the main bus. After the diesel generator is started and comes up to speed, a switch closes, connecting the diesel generator to the main electrical bus. The loads would then be sequenced back on, either manually or automatically. If the diesel generator fails to supply the air-conditioning power bus, the battery remains to supply instruments and controls.

2.1.2.6 Heating, Ventilation, and Air-Conditioning System

A heating, ventilation, and air-conditioning system can be designed in many different ways to accomplish a particular purpose. Codes specify certain general configurations, but a designer has latitude when configuring the ducts, fans, and dampers. Here, the exhaust side has been assumed to consist of three trains: two normally operating trains and a standby train. When radioactive material is detected, the normal train isolates the contaminated area. It is assumed that the system operates continuously.

Based on the discussion presented in Section 2.1.1.6, the heating, ventilation, and air-conditioning system associated with the spent fuel pool at the conceptual Wet Handling Facility will be assumed to provide for flow of air from areas of low potential for radioactive

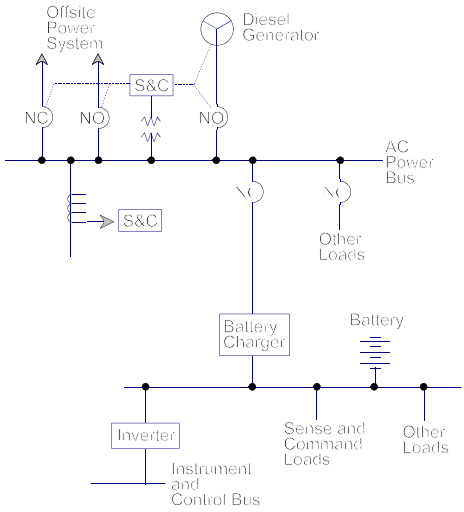


Figure 2-3. Electrical Diagram (NC: Normally Closed; NO: Normally Open; S & C: Sense and Command)

contamination to areas of higher potential for radioactive contamination, as indicated in ANSI/ANS—57.7—1988. For this purpose, the conceptual Wet Handling Facility will be assumed to be separated into three zones: (i) primary confinement, (ii) secondary confinement and (iii) tertiary confinement. The primary confinement zone is the work area that can become contaminated, since bare fuel assemblies are handled in the spent fuel pool and load casks and canisters are lifted during normal operations. The secondary confinement zone includes the areas that could potentially become contaminated if the heating, ventilation, and air-conditioning system fails to maintain negative differential pressure (i.e., areas adjacent to the spent fuel pool room). The tertiary confinement zone is the uncontaminated area (e.g., administrative areas). Each confinement zone is assumed to have its own heating, ventilation, and air-conditioning system. There is a pressure gradient along these three confinement zones: the tertiary confinement zone has the highest pressure, and the primary confinement zone has the lowest pressure so that, in case of contamination, the contaminated area would not leak out of the primary confinement zone.

The exhaust side draws more air than the supply side delivers to maintain a slightly negative pressure in the work area (the difference in the flow is made up by leakage into the work area). The system is assumed to be designed so that the negative differential pressure is maintained when a fan trips. A bank of fans, instead of a single fan, may move air through either side of the heating, ventilation, and air-conditioning system. For example, a bank may consist of three fans: two of which are operating and one that is in a standby mode. Each of the operating fans moves 50 percent of the air that is needed. Hence, it is assumed that two exhaust fans must be operating for the supply side to operate. If one fan trips, the standby fan comes online to maintain the necessary flow. The changeover to a standby train is assumed to be manual in this case (it can be either manual or automatic).

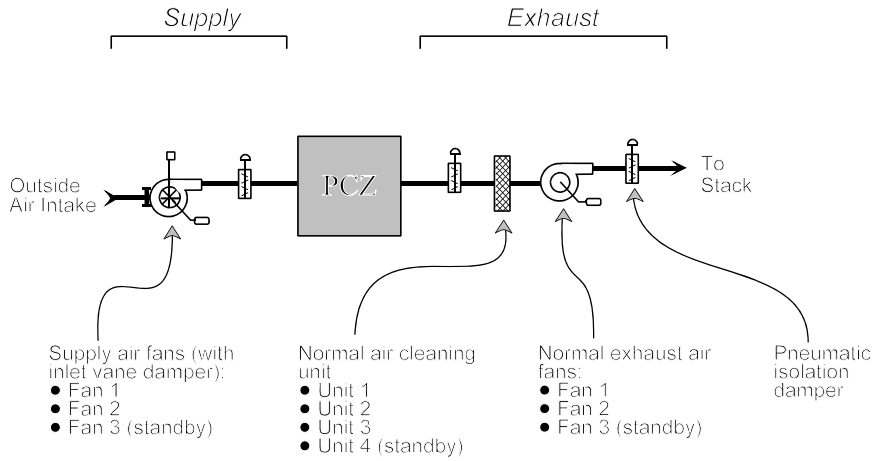


Figure 2-4. Heating, Ventilation, and Air-Conditioning System (PCZ: Primary Confinement Zone)

The heating, ventilation, and air-conditioning system for the primary confinement zone will be continuously operated, regardless of whether airborne radioactive material is present. Figure 2-4 illustrates a plausible configuration of the heating, ventilation, and air-conditioning system for the conceptual Wet Handling Facility. The system consists of a supply side and an exhaust side. Each side consists of three fans: two that are operating and one that is a standby. The standby train starts to maintain the negative pressure in the primary confinement zone only if a failure occurs in one of the normally operating trains. Each fan has an inlet and outlet damper. Presumably, if a damper on a given fan closes while a fan is operating, the fan will trip because of low flow and high temperature.

The high efficiency particulate air filters remove particulates from the airstream. In a previous design where fuel was handled in air, a remote filter would be located on the exhaust side of the primary confinement zone to reduce contamination along the ducts leading to the filter bank similar to those in nuclear power plants. The conceptual Wet Handling Facility is expected to be inherently cleaner due to this design change. Therefore, for the conceptual design, no remote filter was assumed.

2.2 Lifting Operations for Conceptual Facility (Task 2)

The routine lifting operations in the conceptual Wet Handling Facility involve lifts of casks and dual purpose canisters; lifts of individual fuel assemblies; the lift and transfer of a loaded transportation, aging, and disposal canister to a welding area; and the lift and transfer of a loaded and welded transportation, aging, and disposal canister into an aging cask for movement out of the facility. Some potential nonroutine lifting operations were also considered, and examples are provided in Section 2.2.2.

2.2.1 Routine Lifting Operations

For the conceptual Wet Handling Facility shown in Figure 2-1, transportation casks are received through the main vestibule in the horizontal position. Impact limiters would be removed from the transportation casks, and upended in the handling area. The cask handling crane (i.e., overhead bridge crane) lifts the transportation cask and transfers it to the preparation area. In the preparation area, the cask would be prepared for fuel unloading (e.g., unbolting the cask top lid or cutting of lid welds, gas sampling). The transportation cask would then be transferred to the spent fuel pool using the cask handling crane and lowered onto the cask staging shelf in the pool (Figure 2-2). From the cask staging shelf, the transportation cask would be lifted again and transferred to the transportation cask unloading area.

In the conceptual spent fuel pool shown in Figure 2-2, individual commercial spent nuclear fuel assemblies would be transferred from a transportation cask or canister into a transportation, aging, and disposal canister underwater, using a spent fuel transfer machine. The transportation, aging, and disposal canister is located inside a shielded transfer cask. After fuel is transferred into a transportation, aging, and disposal canister, the entire assembly—consisting of the transportation, aging, and disposal canister inside the shielded transfer cask—would be lifted from the transportation, aging, and disposal loading area using the cask handling crane and placed on the cask staging shelf. Before lifting, the transportation, aging, and disposal canister would have a lid in place (but not welded) and the shielded transfer cask lid would be bolted.

The cask handling crane then would lift the shielded transfer cask carrying the transportation, aging, and disposal canister from the cask staging shelf in the pool up and over the edge of the pool. The shielded transfer cask would be decontaminated and the transportation, aging, and disposal canister/shielded transfer cask would then be transferred to the welding area. The shielded transfer cask lid would be unbolted and lifted from the shielded transfer cask. Welding would be performed on the lid(s) for the transportation, aging, and disposal canister. After welding, the lid is returned to the shielded transfer cask and bolted again. The shielded transfer cask with the transportation, aging, and disposal canister inside would then be lifted using the cask handling crane onto a trolley, which moves it into the transfer area. The canister transfer machine in the transfer area unbolts and removes the lid from the shielded transfer cask; lifts the transportation, aging, and disposal canister from the shielded transfer cask; and places the canister into an aging overpack.

The side vestibule is also used to receive dual purpose canisters from the aging facility. The aging overpack containing a dual purpose canister would be moved into the transfer area through the side vestibule. The aging overpack lid would be unbolted, and the canister transfer machine in the transfer area would transfer the dual purpose canister into a shielded transfer cask already positioned on a trolley.

2.2.2 Nonroutine Lifting Operations

In addition to the routine lifting operations, some nonroutine lifting operations involving the spent fuel pool may also take place in the conceptual Wet Handling Facility. These would include:

- Lifting portable filtration or cleaning equipment into the pool
- Lifting and repositioning equipment (e.g., frames) that is in the pool
- Lifting pieces that may have broken off a fuel assembly
- Lifting canisters for remediation purposes
- Lifting and repositioning cameras for improved viewing of underwater operations in the pool

3 OPERATIONAL HAZARD ANALYSIS AND IDENTIFICATION OF INITIATING EVENTS (TASK 3)

This section identifies operational hazards and initiating events at a conceptual Wet Handling Facility described in Section 2.1.2. Hazard analysis examines a wide spectrum of potential events that may cause radiation exposure to public and/or workers. A systematic analysis of natural, human-induced, and operational hazards is an important and initial step in a preclosure safety analysis. In the PCSA Phase II Exercise, however, the scope of identification of hazards and initiating events that may lead to potential Category 1 or 2 event sequences is limited to the analysis of operations in the spent fuel pool only.

3.1 Hazard Analysis

Operational hazards can result either from component or system failure, human error, or a combination of both during fuel handling operations. There are several well recognized methods for conducting a hazard analysis (e.g., what-if method, failure mode and effect analysis, check list) (Kumamoto and Henley, 1996). In this exercise, hazards were identified based on the staff review of information from NUREGs, standards, and regulatory guides on spent fuel pool operations at nuclear power plants as discussed in Section 2.1.1. The review collected information on potential hazards from previous risk analyses and past events from operating plants. The following possible hazards relevant to the conceptual Wet Handling Facility at the Geologic Repository Operations Area were considered.

- Spent nuclear fuel in the pool is exposed to air due to low water level
- Spent nuclear fuel release due to a fuel assembly or cask drop
- Potential criticality due to misload of transportation, aging, and disposal canisters

Scenarios that could create these hazards include (i) loss of spent nuclear fuel pool water cooling; (ii) loss of spent nuclear fuel pool water; (iii) drop of a transportation, aging, and disposal canister containing fuel assemblies leading to a damaged fuel scenario; and (iv) transportation, aging, and disposal canister misload during loading of spent fuel assemblies. The scenarios were selected from a range of possible internal events based on literature reviews and team experience. Effects from any natural and human-induced hazards, such as seismic events and aircraft crashes, are not included, because these hazards were outside the scope of this exercise.

3.2 Initiating Events

After the hazards and exposure scenarios were identified, initiating events that may lead to radiological consequences were examined. Potential initiating events are examined for the following scenarios considered in this exercise: (i) loss of pool water cooling, (ii) loss of pool water, (iii) fuel damage, and (iv) transportation, aging, and disposal canister misload.

3.2.1 Loss of Cooling to the Pool Water

As discussed in Section 2.1.2.3, the conceptual spent nuclear fuel pool is equipped with a cooling system to remove heat from the spent nuclear fuel pool water to maintain water temperature at a preset range. When the pool water cooling function is lost, water may heat up

and eventually boil from the heat generated by the spent nuclear fuel. The loss of cooling of the pool water may result from component failure in the spent fuel pool cooling system or human error during the operation (NRC, 2001, 1997a). The cooling system consists of various mechanical components (e.g., pumps, heat exchanger) that are powered by electricity and controlled by an electronic instrumentation and control unit. Potential initiating events that may lead to the loss of pool water cooling are listed in Table 3-1.

The consequence of loss of pool water cooling was assessed by comparing the risk at nuclear power plant pools for boiling water reactors and pressurized water reactors with the conceptual Wet Handling Pool. The pool characteristics (e.g., pool dimensions and volume, pool operating temperature, and maximum number of assemblies handled in the pool, etc.) for nuclear power plants and for the conceptual Wet Handling Facility are given in Table 3-2. For nuclear power plants, pool data are obtained from NUREG–1738, while data for the conceptual pool are based on publicly available information (Tooker and Dunn, 2006). NUREG–1738 documented a thermal-hydraulic assessment of the spent fuel pool in nuclear power plants to determine the time available for the operators to take actions to prevent zirconium fire caused by uncovery of fuel. The time required to heat the pool water to boiling followed by pool water level dropping down to within 0.9 m [3 ft] of the top of the spent fuel from the normal pool level due to boiling was calculated for fuels with different ages. The boil off time for 5-year-old fuel is estimated to be about 2 weeks (400 hours for pressurized water reactors and 459 hours for boiling water reactors, respectively) (NRC, 2001, Table 2-1).

As shown in Table 3-2, the conceptual pool holds about 6 to 10 times more water compared to a nuclear power plant spent nuclear fuel pool. Additionally, the conceptual pool stores about half as many fuel assemblies as pools in power plants for boiling water reactors and one tenth the number in pressurized water reactor pools. The age of the spent nuclear fuel to be handled at this conceptual pool is assumed to be at least 5 years old. Qualitatively, it can be assumed that the conceptual pool will take more than 2 weeks to boil to within 0.9 m [3 ft] above the spent fuel and it is expected that operators will have sufficient time to take corrective actions upon loss of spent nuclear fuel pool water cooling to restore the cooling capability. Hence, the initiating events that may cause the loss of pool water cooling were not further developed in this exercise, because of the slow evolution of the event and long lead time to take corrective actions.

3.2.2 Loss of Pool Water

This scenario is similar to the loss of pool water cooling scenario in Section 3.2.1 with one major difference: the previous case deals with loss of water caused by water boil off, whereas this scenario is characterized by an uncontrolled loss of pool water from the spent fuel pool. The loss of pool water may result from damage to the spent nuclear fuel pool wall or floor or siphoning of the pool water (NRC, 2001, 1997a). Potential initiating events that may lead to the loss of pool water are listed in Table 3-3.

The spent fuel pool at the conceptual Wet Handling Facility will be below ground level and is assumed to consist of thick walls and a floor slab designed to appropriate load combinations, such as seismic, to provide leak-tight integrity. If drop of an empty or loaded transportation or

Table 3-1. Potential Initiating Events That May Lead to the Potential Loss of Cooling of Pool Water	
Initiating Event	**Description**
1. Component failure in normal and standby trains of the cooling system	A myriad of mechanical components (e.g., a heat exchanger or piping) in the cooling system may fail, leading to the loss of the cooling system. For example, if a pump fails to continue running in the operating train and a valve fails to open in the standby train, then cooling would be lost to the spent fuel pool.
2. Electrical system failure	Failure of the normal and backup power supplies would cause loss of power to the cooling system pumps, leading to insufficient recirculation of cooling water.
3. Control system failure	A failure in the control system that controls the modulation or opening/closing of a valve or software errors may cause portions of the cooling system to become inoperative.
4. Valve misalignment	Cooling may be lost due to valve misalignments at the heat exchangers. For instance, valves could be misaligned due to human error, thereby blocking the flow of cooling water.

Table 3-2. Pool Conditions for Different Spent Fuel Pools

Facility	Fuel Type	Surface Area, m^2 [sq ft]	Depth of Water, m [ft]	Pool Volume, m^3 [cu ft]	Initial Water Temp, °C [°F]	Number of Fuel Assemblies
Nuclear Power Plant	Boiling Water Reactor*	105.7 [1,137.7]	11.54 [37.86]	1222 [43,155]	30 [86]	4,200 (9 × 9)
	Pressurized Water Reactor*	61.3 [659.8]	11.54 [37.86]	707 [24,967]	30 [86]	965 (17 × 17)
Conceptual Wet Handling Facility	Pressurized Water Reactor and Boiling Water Reactor†	425 [4574.7]	15.9 [52.17]	6758 [238,656]	32 [90]	Assumed 400

*NRC. NUREG–1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants." Washington, DC: NRC. 2001.
†Tooker, D. and T. Dunn. "Surface Facilities—Initial Handling Facility, Wet Handling Facility, and Receipt Facility, NRC/DOE Technical Exchange and Management Meeting on Design Changes Approved Through DOE's Critical Decision-1 (CD-1) Process." Bechtel SAIC Company, LLC. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2006.

Table 3-3. Initiating Events for Loss of Pool Water	
Initiating Event	**Description**
1. Transportation cask drop*†	Unbolted shipping cask containing fuel assemblies and dual purpose canister is dropped in the pool while being transferred from cask preparation and dual purpose canister cutting area to the pool. The floor or wall may crack from a drop impact, causing water to leak from the pool.
2. Site specific cask drop	Site-specific cask with unsealed transportation, aging, and disposal canister is dropped in the pool while being lifted from the transportation, aging, and disposal loading area, cracking the floor or wall of the pool.
3. Water is siphoned from the spent fuel pool	Water may be inadvertently siphoned from the spent fuel pool due to human errors. For instance, a path may be accidentally established to siphon water out of the pool in a maintenance evolution.

*NRC. NUREG/CR–4982, "Severe accidents in spent fuel pools in support of Generic Issue 82." Washington, DC: NRC. 1987.
†_____. NUREG–1353, "Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools." Washington, DC: NRC. 1989.

site-specific casks is considered to be a credible event, the pool design load combinations and design should include an impact load from a drop of cask or canister on a pool floor and walls. Consequently, it is assumed that a cask drop would not sufficiently damage the pool structure so that a rapid drawdown of pool water would uncover spent nuclear fuel. It is also assumed that leak detection is part of the instrumentation and control system in the conceptual Wet Handling Facility for sensing depletion of the pool water level and taking corrective actions. Hence, the initiating events that may cause loss of pool water from dropping a transportation and a site-specific cask are screened out and were not further analyzed in this exercise.

The loss of water due to siphoning in nuclear power plant spent fuel pools is mitigated by the system design and operating procedures (NRC 2001, 1997a), which requires that (i) a source of emergency makeup water available that remotely supplies water is available, (ii) portable pumps are available, (iii) radiation levels and coolant levels are monitored from a control room, (iv) heavy lifts over the spent fuel pool wall (i.e., having a design that uses gates or weirs) are prevented, and (v) design, procedures, and administrative controls are in place to ensure antisiphon protection is available. Provisions similar to those in nuclear power plant pools are assumed to exist in the conceptual pool and, consequently, potential siphoning of pool water will be mitigated. Hence, the initiating events that may cause loss of pool water because of siphoning were not pursued further and were screened out in this exercise.

3.2.3 Fuel Damage

The fuel may be damaged by dropping of a fuel assembly during loading/unloading operations in the pool or by dropping a piece of equipment onto a fuel assembly during these operations. Potential fuel damage scenarios and the associated initiating events are shown in Table 3-4.

Initiating Events 1 through 8 occur under water. The water above the fuel assemblies provides effective scrubbing of any fuel fines that may be released following a fuel damaging incident. A radiological dose to the workers from these eight initiating events, if any, will be bounded by the worker dose from dropping an unsealed transportation, aging, and disposal canister (inside shielded transfer cask) onto a concrete floor outside the pool (initiating Event 9) because the radioactive contaminants will be released in the air. Therefore, these eight initiating events are not analyzed further in this exercise. Initiating Event 9 is the drop of a transportation, aging, and disposal canister onto the concrete floor. The event sequence resulting from this initiating event will be further developed in Section 4.1.

3.2.4 Transportation, Aging, and Disposal Canister Misload

Misload of a transportation, aging, and disposal canister is characterized by improper loading of spent fuel assemblies in the canister during handling operations or a neutron-absorber plate is missing in a transportation, aging, and disposal canister due to manufacturing error.

The first scenario can be precluded due to the different dimensions of the fuel assemblies. Typically, a boiling water reactor assembly is 4.2 m [14 ft] long and a pressurized water reactor assembly is 3.6 m [12 ft] long. The pressurized water reactor assembly is wider than a boiling water reactor assembly. Furthermore, the accessories (e.g., clamps, pads, and other restraints) on a pressurized water reactor assembly will prevent it from entering into a boiling water reactor canister because the accessories will be in wrong positions (Baker, 2007). Similarly, a boiling water reactor fuel assembly will not be able to fit into a pressurized water reactor canister.

A missing or incorrectly positioned neutron-absorber plate could potentially lead to criticality. Preliminary calculations performed by the NRC staff (Whaley, 2007) suggested that the cask would remain subcritical even with four boral neutron absorber plates missing from HI-STORM MPC–24 (Whaley, 2007). More analyses are needed to investigate the neutronic reactivity effect of missing neutron-absorber plates from transportation, aging, and disposal canisters; these are outside the revised scope of this exercise.

Table 3-4. Initiating Events That May Lead to Fuel Damage Scenarios	
Initiating Event	**Description**
1. Drop of a fuel assembly due to spent fuel transfer machine equipment failure (including control system failure and human errors).	The spent fuel transfer machine drops a fuel assembly onto a storage rack or the spent nuclear fuel pool floor or a transportation, aging, and disposal canister.
2. Fuel assembly is inadvertently lowered onto another fuel assembly due to human error while operating the spent fuel transfer machine.	A fuel assembly is inadvertently lowered onto another fuel assembly in either the storage rack or the transportation, aging, and disposal canister or onto the storage rack itself or the transportation, aging, and disposal canister itself.
3. Fuel assembly becomes stuck or damaged while being raised or lowered due to spent fuel transfer machine lateral movement during raising or lowering operations.	Spent fuel transfer machine moves laterally while a fuel assembly is being raised or lowered resulting in the fuel assembly being bent or broken and the cladding damaged.
4. Fuel contained in a dual purpose canister or transportation cask was damaged during or prior to shipment.	Upon visual inspection of the fuel in a dual purpose canister or transportation cask, it is discovered that fuel has been damaged.
5. Dual purpose canister lid is dropped onto loaded dual purpose canister following cutting operations.	Following cutting operations, the lid of the dual purpose canister is dropped onto the dual purpose canister at an appropriate location, thereby causing damage to fuel assemblies.
6. Drop of a lid onto a loaded transportation, aging, and disposal canister or loaded transportation cask.	Lid is inadvertently dropped onto a loaded transportation, aging, and disposal canister or loaded transportation cask at an appropriate location, thereby damaging fuel assemblies.
7. Spent fuel transfer machine lifting mechanism strikes a fuel assembly in either the storage rack; a cask; the transportation, aging, and disposal canister; or a dual purpose canister.	Different sized casks would require a different height of lift for the spent fuel transfer machine. A fuel assembly may be damaged if the incorrect lifting operation is used. For example, the Spent fuel transfer machine hoist may be commanded to lower too far. It may lower and strike a fuel assembly.
8. Component of the spent fuel transfer machine drops onto a fuel assembly in either the storage rack; a cask; the transportation, aging, and disposal canister; or a dual purpose canister.	Component of the spent fuel transfer machine falls and strikes a fuel assembly. For example, a grapple may inadvertently fall from the spent fuel transfer machine and strike a fuel assembly.
9. A loaded transportation, aging, and disposal canister inside a shielded transfer cask is dropped on a concrete floor outside the pool.	After fuel is loaded into a transportation, aging, and disposal canister, shielded transfer cask is lifted from the spent fuel pool and transferred to the transportation, aging, and disposal welding area. During the transfer, the shielded transfer cask can be dropped due to a crane hardware failure or human errors. If the transportation, aging, and disposal canister is dropped during this operation, some of the fuel inside the canister may become damaged.

4 EVENT SEQUENCE ANALYSIS FOR A CASK DROP

4.1 Development of Event Sequence

4.1.1 Initiating Event Description

The event sequences were developed based on the scenario of dropping a loaded cask onto a hard surface. The two drop scenarios identified in NUREG–1864 (NRC, 2006) are two-blocking and failure of a flawed cable. Two-blocking is defined as the continued hoisting to the extent that the upper head block and the load block are brought into contact. The excessive loads created in such an event may lead to rope failure and dropping of the load (NRC, 1980). Other possible causes of load drop include load hang-up, which is the act of a load block being stopped by a fixed object during hoisting, thereby overloading the hoisting system (NRC, 1980). In general, the majority of crane accidents (including dropping a load) can be attributed to human actions (NRC, 2003a). Examples of general human performance problems that could contribute to cask drops include, but are not limited to, deficient procedures, imperfect communication between the operators, and complacency as a result of repeated actions.

4.1.2 Event Sequence Model

This section describes the event sequence model of the scenario described under Section 4.1.1. The initiating event is a heavy load drop due to a crane failure while lifting a transportation, aging, and disposal canister inside a shielded transfer cask (according to the description provided in Section 3.2). The frequency of the initiating event will be estimated in Section 4.2. The subsequent events for the event sequence analysis due to the heavy load drop are (i) impact of the shielded transfer cask onto the concrete floor; (ii) potential damage of the fuel assemblies inside the transportation, aging, and disposal canister; (iii) release of noble gases from the unsealed transportation, aging, and disposal canister; and (iv) failure of the heating, ventilation, and air-conditioning system to exhaust gas releases from the primary confinement.

For consequence calculation purposes (to be further discussed in Chapter 5), the floor workers may be exposed to noble gas release for an extended period of time for which a conservative consequence assumption would result if heating, ventilation, and air-conditioning failure to exhaust is assumed. If the heating, ventilation, and air-conditioning successfully exhausts the releases to the outside air (through the exhaust stack), a dose for the public and outside workers may potentially occur. Given the complexity in quantifying the probability of fuel damage (and subsequent release given the shielded transfer cask and the transportation, aging, and disposal canister configuration), it is assumed for simplicity of this analysis that the probability of occurrence of this event is 1. While this is a conservative assumption, it allows the event sequence frequencies to be estimated with the remaining events in the event tree. This event is established as the failure of the heating, ventilation, and air-conditioning system to exhaust the releases and, therefore, requires quantification to perform the posterior categorization (including uncertainty and sensitivity analysis). The resulting event tree model based on these assumptions is shown in Figure 4-1. The indicated probabilities P_1 and P_2 in

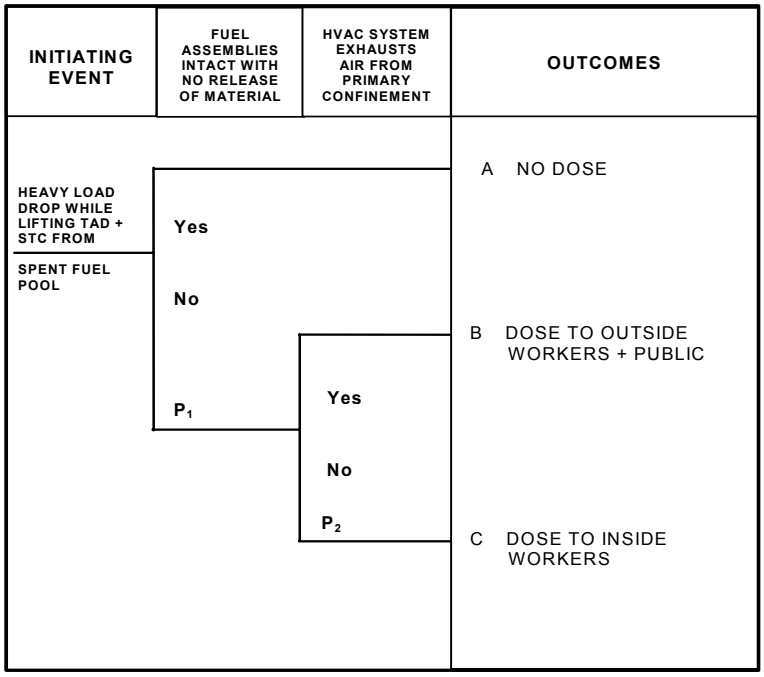


Figure 4-1. Event Tree Model for Drop Scenario (TAD: Transportation, Aging, and Disposal; STC: Shielded Transfer Cask; HVAC: Heating, Ventilation, and Air Conditioning)

the developed event tree branches correspond to the fuel assembly damage and release of noble gases given a drop has occurred (assumed to be 1) and the failure of the heating, ventilation, and air conditioning to exhaust the air in the primary confinement zone (to be estimated in subsequent sections). Event Sequence A results in no dose, given that releases are assumed to be contained within the transportation, aging, and disposal canister. Event Sequence B considers the consequences to the outside worker and public; while Event Sequence C may result in the more conservative dose calculation to inside workers. Other considerations that can be modeled using this approach (but that are not included for the purposes of this PCSA Phase II Exercise) are (i) failure to maintain negative pressure gradient, resulting in potential doses to adjacent rooms that may also be occupied by workers (i.e., control room operators); (ii) failure of alarms and radiation detector devices to alert inside workers to the release of noble gases; and (iii) failure of the indoor workers to follow emergency procedures during accident events. It is anticipated that dose due to the first alternative scenario postulated above may be significantly lower to control room operators than to the floor workers closer to the area where the drop occurs. Therefore, the event tree model shown in Figure 4-1 was chosen for quantification purposes.

4.2 Reliability Estimation

4.2.1 Crane Reliability

Two approaches for estimating the drop probability are to perform a reliability analysis of the crane used to lift the transfer cask and to obtain an empirical estimate based on experience with similar types of cranes lifting heavy loads. Although the first approach may provide more insight, it is much more complex than the second approach. A reliability model must account for (i) lifting equipment (e.g., crane, yoke); (ii) lifting operations; and (iii) maintenance. While the reliability of the equipment itself may be more or less straightforward, accounting for human actions and maintenance is a significant undertaking.

The probability of dropping the transfer cask can be estimated from data on lifts of very heavy loads at U.S. nuclear power plants. The data used in NUREG–1774 (NRC, 2003a) were evaluated for use in this study. The application of NUREG–1774 is not a matter of having identical plants, but having similar relevant aspects of lifting a cask so these plants can be analyzed in the same way. In this exercise, we assume that the relevant aspects of cask-lifting at the hypothetical wet-handling facility would be sufficiently similar to justify using the NUREG-1774 estimates.

NUREG–1774 describes the results of a survey of crane operating experience at U.S. nuclear power plants during 1968–2002. Nine sites were visited to collect operational data on very heavy load lifts. Each site was chosen to represent one type of plant (e.g., nuclear steam supply system). Although some sites were selected for convenience, the nine sites are broadly representative of all nuclear power plant sites in the United States. The nine selected sites included 19 plants. The number of very heavy load lifts was determined from maintenance logs, then extrapolated to the remaining power plants in the United States. The total number of heavy lifts for all operating plants in United States was determined to be 54,000.

The number of times that drops occurred was obtained from docketed information and licensee event reports of all plants in the United States (not just the representative plants). In the subject time period (1968–2002), three drop events were determined to be relevant to operations similar to those assumed for this conceptualized Wet Handing Facility. The probability of dropping a very heavy load is determined from Eq. (1).

$$Pr(\text{drop}) = \frac{N_{\text{drops}}}{L_{\text{total}}} \tag{1}$$

where

N_{drops} = number of drops of a very heavy load
L_{total} = number of lifts of a very heavy load

Solving Eq. (1) with $N_{\text{drops}} = 3$ and $L_{\text{total}} = 54{,}000$, the probability of dropping a cask given a lift is 5.6×10^{-5}. The frequency of dropping a cask is given by Eq. (2).

$$f(\text{drop}) = f(\text{lift}) \times Pr(\text{drop}) \quad (2)$$

where

$f(\text{drop})$	=	frequency of dropping a cask
$f(\text{lift})$	=	frequency of lifting a cask

From experience in dry cask storage operations, it will take about 3 to 4 days total to unload a dual purpose canister, and load and dry the transportation, aging, and disposal canister. Therefore, about 100 transportation, aging, and disposal canisters could be loaded in a year. Consequently, the frequency of lifting transportation, aging, and disposal canisters out of the pool is assumed to be 100 per year. Substituting the probability of dropping a very heavy load from Eq. (1) and the lift frequency from operating experience into Eq. (2) yields a drop frequency of 5.6×10^{-3} per year.

4.2.2 Heating, Ventilation, and Air-Conditioning System Reliability

Based on the scenario described in Section 4.1.1 and the event sequence modeling assumptions presented in Section 4.1.2, a reliability estimate is developed in this section for the heating, ventilation, and air-conditioning system conceptualized in Section 2.1.2.6.

The conceptual model of the heating, ventilation, and air-conditioning system is composed of a supply train, a filter bank, and an exhaust train that operate continuously (Section 2.1.2.6). Both supply and exhaust functions are performed by a 2-out-of-3 train configuration, where two operating trains must be running to maintain the heating, ventilation, and air-conditioning system functions of (i) filtering any potential release of airborne particles, (ii) exhausting potential gas and particulate releases from the primary confinement zone to the outside air, and (iii) maintaining negative differential pressure within the facility. In this section, it is also assumed that the standby train in both the supply and exhaust portions of the heating, ventilation, and air-conditioning system must be manually activated by a human operator to maintain these functions above.

For the scenario described in Section 4.1.1, a high efficiency particulate air filter was not considered for reliability estimation purposes. Also, based on the event sequence assumptions, the failure to exhaust any releases resulting from a drop event inside the conceptual Wet Handling Facility is identified as the potential heating, ventilation, and air-conditioning system failure mode (i.e., top event) under consideration. Hence, this analysis assumes that the heating, ventilation, and air-conditioning exhaust trains are the critical portion of the system to be analyzed for estimating the probability of occurrence of the top event. Figure 4-2 shows the simplified exhaust model with the 2-out-of-3 configuration that assumes three identical parallel trains with a fan and an isolation damper in each train.

With the previously noted considerations (Section 4.1.2), a fault tree analysis of the simplified conceptual heating, ventilation, and air-conditioning exhaust system in Figure 4-2 is implemented. Additional assumptions include (i) only human action can produce a failure to

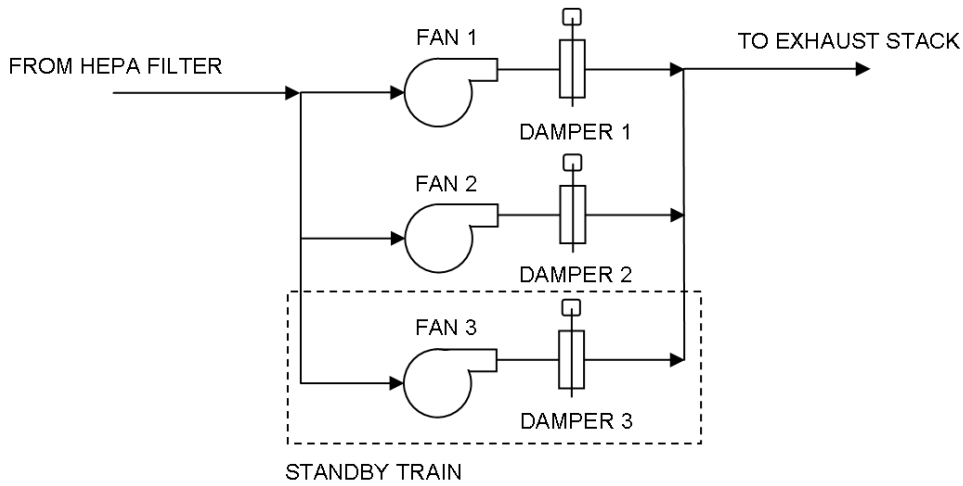


Figure 4-2. Heating, Ventilation, and Air-Conditioning System Exhaust Configuration for Reliability Analysis

switch between an operating train and the standby train (i.e., zero probability of failure to switch due to mechanical/electrical issues); (ii) switching time is negligible compared to the system running time; (iii) failure of the electrical supply distribution system to support the heating, ventilation, and air-conditioning functions is not considered for simplification of the analysis; (iv) common-cause failure (i.e., dependent failure affecting all trains in the exhaust model) is assumed not to be possible; and (v) all dampers and fans in either operating or standby trains are assumed to be provided by the same vendor. Although not modeled here, both the failures resulting from loss of electrical supply to the heating, ventilation, and air-conditioning system and common-cause failures can be included within the same framework of the fault tree analysis presented here (Kumamoto and Henley, 1996).

The fault tree developed under these considerations is shown in Figure 4-3, where the top event can occur if either Train 1 or Train 2 fails to run and Train 3 (in standby mode) is not activated (i.e., fails to start or run). At the train level, failure of either Train 1 or Train 2 can occur if Fan 1 or Fan 2 fail to run or if Damper 1 or Damper 2 fail closed. For mechanical failure of the standby train to occur, either Fan 3 must fail to start or run or the damper must fail to open or remain open. Additionally, Train 3 can fail if the undeveloped basic event of the human operator failing to activate the standby train occurs.

The main input parameters for the fault tree analysis are the failure rates and probabilities of failure on demand for the dampers and fans, the failure on demand of the human operator, and the mission time and the assumed number of hours the standby components must run if a failure occurs in one of the operating trains. Failure data was selected from a generic database (Blanton and Eide, 1993), which provides both failure rates and probabilities of failure on demand for a variety of system components. This includes information on the statistical distribution and spread on the rates and probabilities that can be used for uncertainty analysis to be performed in Section 4.3. These values are presented in Table 4-1. For the basic event caused by undeveloped human operator failure, screening type failure probability

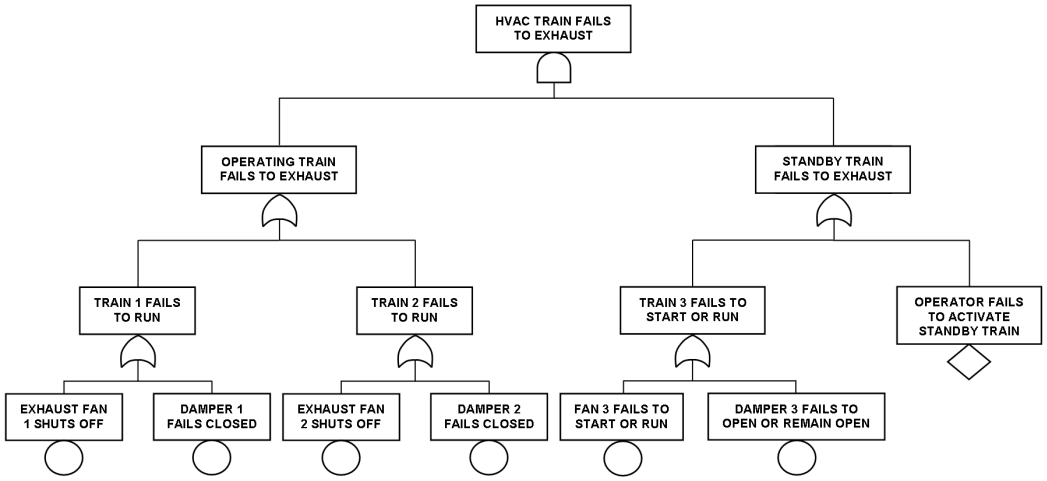


Figure 4-3. Fault Tree for the Heating, Ventilation, and Air-Conditioning Train Failure to Exhaust

values can be assumed for the purposes of the fault tree top event quantification, along with sensitivity studies that address how different point estimates can be included to represent various operational scenarios and contexts. After defining the input parameters, the fault tree presented in Figure 4-3 can be solved using the minimal cut set analysis (Kumamoto and Henley, 1996; NRC, 1981).

This type of analysis is particularly useful when large fault trees are developed for a complex system with multiple failure modes. Although the fault tree considered here is relatively simple, using the minimal cut set analysis showcases how a larger fault tree may be solved if additional failure modes (e.g., loss of electric power supply to heating, ventilation, and air conditioning, control equipment failure; additional human operator errors) were to be included. Because the fault tree analysis includes the potential for standby failures, specific reliability parameters of interest are considered (AIChE, 2000): (i) the unavailability (i.e., the probability that a failure exists at a specified time), (ii) the unreliability (i.e., the probability that a failure occurs during a specified time interval), and (iii) the undependability (i.e., the probability that the event exists at a specified time or occurs during a given time interval). Undependability is a suitable reliability parameter for standby systems, because failure modes include failure to start or failure to run the standby components for a required amount of time. Ultimately, the probability of the top event is calculated as the unreliability of the heating, ventilation, and air-conditioning system with a 2-out-of-3 train configuration to exhaust the releases from the primary confinement zone in a given interval of time.

To quantify the unreliability of the top event, the unreliabilities of the individual minimal cut sets must be calculated first. For the fault tree shown in Figure 4-3, 12 minimal cut sets can be identified (i.e., the minimal combination of basic events that results in the system failure mode).

Table 4-1. Failure Data for Individual Components		
Component	Failure to Operate or Run (1/hour)	Failure on Demand
Damper	3.0×10^{-6}	3.0×10^{-2}
Fan	3.0×10^{-5}	5.0×10^{-3}
*Blanton, C.H. and S.A. Eide. "Savannah River Site Generic Base Development." WSRC–TR–93–262, Rev 1. Aiken, South Carolina: WSRC. 1998.		

For example, the failure of Fan 1 in combination with the human operator failure to activate the standby train produces the top event. Because the fault tree in Figure 4-3 is mainly composed of OR-gates, all the minimal cut sets derived from this analysis are two-event minimal cut sets (i.e., combinations of two basic events) such as in the example provided. For each of these cut sets, the initiating basic event is a component failure in the operating trains (either Train 1 or 2) followed by either a component failure or a human operator failure affecting the standby train.

As an example, the unreliability of the minimal cut set described above can be considered. The failure rate for the fan in Table 4-1 is given as 3.0×10^{-5}/hour (constant failure rates for all components are assumed throughout this analysis). The undependability of the human failure basic event can be assumed to have a value of 10^{-2} (Cooper, 2007) for quantification purposes. The product of these two values results in a rate of failure for this minimal cut set of 3.0×10^{-7}/hour. Assuming the heating, ventilation, and air-conditioning system is maintained and inspected weekly (i.e., a time interval of 168 hours), the approximate unreliability is 5.0×10^{-5}.

In the case of standby component failure, the probability of failure on demand (i.e., the second column of values in Table 4-1) and the failure to run during the time needed for the standby train to maintain the heating, ventilation, and air-conditioning function established above need to be considered. With a minimal cut set consisting of failure of Fan 1 and Damper 3 and assuming a 24-hour standby train mission time for quantification purposes, the probability of failure to remain open for Damper 3 is $(3.0 \times 10^{-6}$/hour$) \times 24$ hours $= 7.2 \times 10^{-5}$. It is assumed that the failure rate for the standby damper is equal to the failure rate for the damper in the operating train (which may not always be valid for all components). Also, independence is assumed between the failure to open and the failure to remain open probabilities, which allows for the summation of this result with the listed failure on demand generic value of 3.0×10^{-2}. This results in an undependability of 0.01 for Damper 3. Following similar calculations for the rate of failure and unreliability of the minimal cut set (also for the 1-week time interval) results in a probability of 1.5×10^{-4}.

Once all the minimal cut set unreliabilities have been calculated, the top event unreliability can be approximated as the summation of the individual values. For the time interval and standby mission times already considered (i.e., 1-week and 24-hour, respectively), the full analysis yields a probability that the top event occurs in the given time interval of 5.1×10^{-4}.

Repeated calculations with different input parameters than those presented previously can be performed (e.g., for sensitivity purposes). The resulting unreliability can then be used as input to the model presented in Section 4.2 for event sequence frequency calculations and subsequent uncertainty analysis (Section 4.3) and/or categorization purposes (Section 4.4).

The assumptions made throughout the calculations reflect considerations that can be useful when estimating an approximate reliability value for systems that may actually be more complex than the one conceptualized here.

4.3 Uncertainty and Sensitivity Analysis

4.3.1 Crane Reliability

Based on the initiating event frequency estimated in Section 4.2.1 and the recently published accepted approach on reliability estimates for a heavy load drop due to crane failure (NRC, 2007a), a sensitivity analysis can be directly applied to the initiating event presented in Section 4.1.1. The drop frequency obtained in Section 4.2.1 directly follows the calculation in NRC (2007a), where 0.167 drops are expected assuming a 30-year operational period. The confidence interval method (NRC, 1994) can be used to estimate the probability that less than one drop occurs during the operational period, which is found to be essentially 100 percent (i.e., there is high confidence that the unmitigated event sequence from a load drop is less than Category 1).

Applying the approach in NUREG–1475 (NRC, 1994, p.18-11) to the data provided in NUREG–1774 (NRC, 2003a), the confidence interval for the frequency of dropping a cask can be obtained as follows. Using the same nomenclature as in NUREG–1475 for clarity, there are $P = 3$ events (i.e., drops) recorded in $t = 54{,}000$ lifts. The degrees of freedom for the lower and upper confidence limits can be calculated as $DF_L = 2 \times P$ and $DF_U = 2 \times (P + 1)$, respectively. For the data considered here, $DF_L = 2 \times 3 = 6$ and $DF_U = 2 \times (3 + 1) = 8$. The lower and the upper confidence limits are given by $\lambda_L = 0.5 \times \chi_{\alpha/2}{}^2(DF_L)$ and $\lambda_U = 0.5 \times \chi^2_{1-\alpha/2}(DF_U)$, respectively, where $\chi^2_q(DF)$ represents the qth quantile for the chi-squared distribution with DF degrees of freedom and $q = 100 \times (1-\alpha)$ percent. For a 95 percent confidence interval, $\alpha = 0.05$ so that the lower and the upper confidence limits can be obtained from $\lambda_L = 0.5 \times \chi^2_{0.025}(6)$ and $\lambda_U = 0.5 \times \chi^2_{0.975}(8)$, respectively. Using the chi-squared built-in function in Mathcad (MathSoft, 1999), the result is $\lambda_L = 1.15 \times 10^{-5}$ and $\lambda_U = 1.62 \times 10^{-4}$. Multiplying these probabilities by the assumed frequency of lifting a cask (as indicated in Section 4.2.1), the corresponding lower and upper limits for the frequency of dropping a cask are 1.15×10^{-3} per year and 1.62×10^{-2} per year. These bounds would result in an expected number of drops of 0.03 and 0.50, respectively (i.e., both are still below the Category 1 limit).

Similarly, the sensitivity analysis in NRC (2007a) illustrates the uncertainties when considering published failure data in reliability estimates. This reference identifies six more drop events (NRC, 2003a) in operational records that could be reasonably considered as additional crane drop failures. If the new total number of drops (i.e., 9) is applied to the analysis in Section 4.2, a crane drop frequency is recalculated as 1.67×10^{-2} per year, with a 95 percent confidence interval defined by a lower limit of 7.62×10^{-3} per year and an upper limit of 3.16×10^{-2} per year. The expected number of drops for a drop frequency of 1.67×10^{-2} per year would now correspond to 0.5 drops. More significantly, for the 95 percent confidence interval upper limit, the expected number of drops would be 0.95. Furthermore, at the 99 percent confidence interval, the expected number of drops during the 30-year operational period would increase to 1.11. Based on the proximity of these values to the categorization limits, additional scrutiny of the assumed crane reliability may be warranted.

4.3.2 Heating, Ventilation, and Air-Conditioning System Reliability

Uncertainty can be introduced in the fault tree analysis presented in Section 4.2 and in the input parameters, such as the failure rates and probabilities of failure-on-demand values in published industry databases (if available). For example, from the generic database developed for the Westinghouse Savannah River Company (Blanton and Eide, 1993), the values associated with the fan and damper failures are assumed to follow a lognormal distribution with an estimated error factor that is the ratio of the 95th percentile value to the median (i.e., 50th percentile) of the lognormal distribution. Values computed for the deterministic fault tree analysis in Section 4.2 are actually listed as the mean of the lognormal distribution assumed to represent the uncertainty in those parameters. Hence, a probabilistic fault tree analysis that considers the uncertainty in input values can be developed based on the statistical information presented in Table 4-1 (Blanton and Eide, 1993).

The corresponding lognormal cumulative distribution functions for both fan and damper are shown in Figure 4-4 for the failure information presented in Table 4-2. With this information, uncertainty analysis can be implemented using, for example, Monte Carlo sampling to propagate the potential range of values indicated in Figure 4-4 through the minimal cut sets. This analysis can be performed with commercially available software, where sampling of random variables can be performed using built-in functions. Assuming the distributions in Table 4-2 and the same mission time for the heating, ventilation, and air-conditioning exhaust system (i.e., 168 hours) and for the standby train given that a failure in an operating train has occurred (i.e., 24 hours), the resulting distribution of the top event probability (i.e., unreliability) is shown in Figure 4-5. This was obtained using 100,000 realizations for each of the parameters in Table 4-2. However, equivalent components (i.e., Fan 1 and Fan 2) were assigned similar values of failure rates, because it was assumed in Section 4.2 that all components were obtained from the same vendor. Hence, for equivalent components, the same failure rate sampled from the distribution was assigned in each realization.

Along with the cumulative distribution function for the actual set of unreliabilities obtained through the propagation of uncertainty, a lognormal fit is included in Figure 4-5. This lognormal distribution is constructed assuming the mean and standard deviation obtained from the simulated results (i.e., 5.1×10^{-4} and 1.0×10^{-3}, respectively) has an error factor of 4.8. The close match between both distributions indicates that the top event could potentially be modeled as a lognormal distribution for this case. For sensitivity analysis, three input parameters can be considered: (i) heating, ventilation, and air-conditioning system mission time; (ii) standby train mission time; and (iii) the undependability of the human operator in switching to the standby train given an operating train failure. The input value for the undeveloped basic event of human operator failure to switch to the standby train has been assumed to be 10^{-1} and 10^{-3} for sensitivity analysis purposes (Cooper, 2007). Given the same mission times stipulated previously (i.e., 168 and 24 hours), the heating, ventilation, and air-conditioning system top event probability results in 1.5×10^{-3} and 4.1×10^{-4}, respectively. Comparing with the previously calculated value in Section 4.2.2 (i.e., 5.1×10^{-4} with a 10^{-2} human operator undependability), an order of magnitude increase occurs if 10^{-1} is used as input. If the standby train mission time is increased from 24 hours to 48 and 72 hours, the resulting failure probability of the heating, ventilation, and air-conditioning system to exhaust remains in the 10^{-4} magnitude range with almost negligible variation (i.e., 5.2×10^{-4} for 48 hours and 5.3×10^{-4} for 72 hours). Similarly, if the mission time of the overall heating, ventilation, and air-conditioning system assumes either a 4-week or 8-week time interval (with a 24-hour standby mission time),

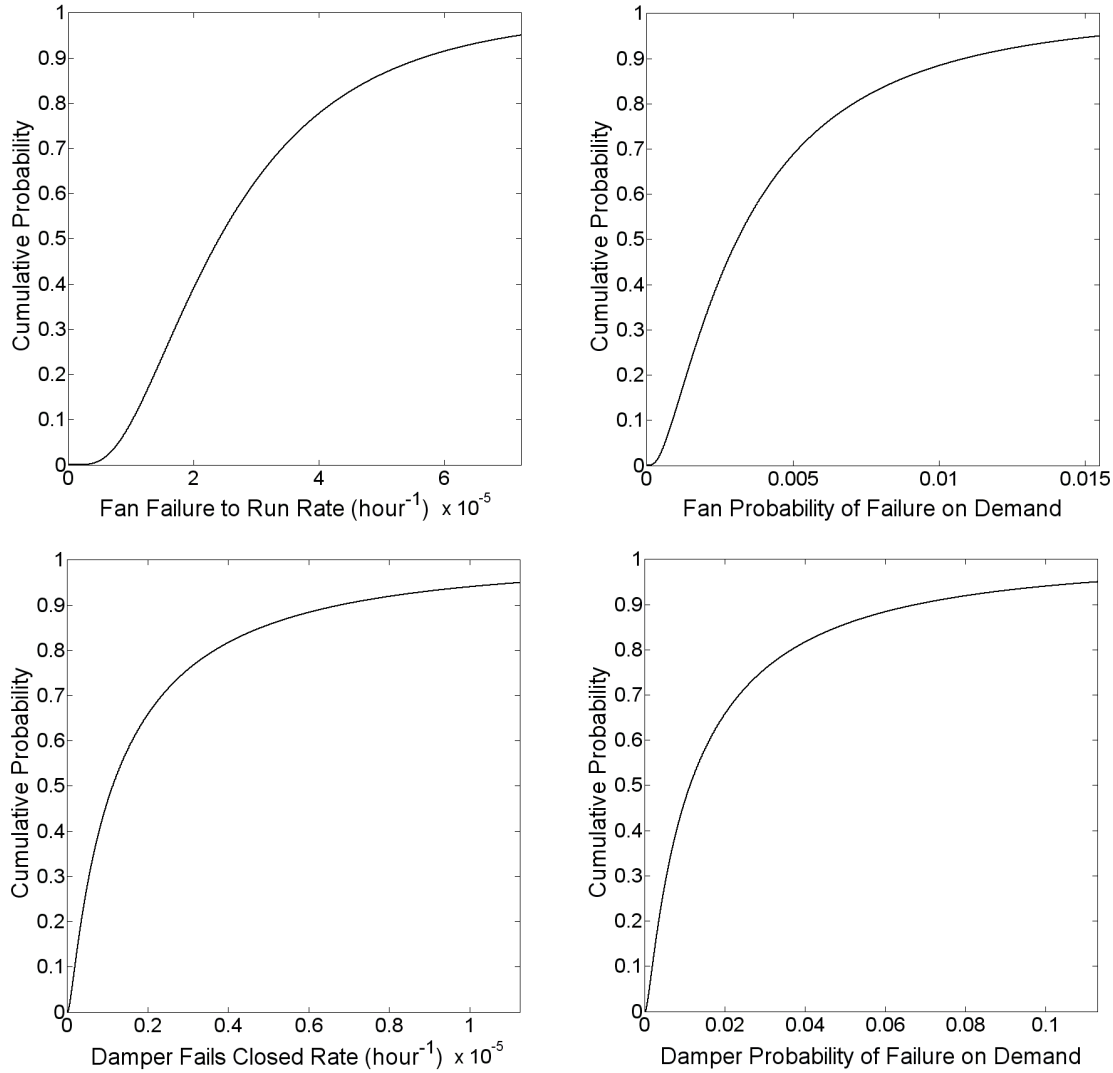


Figure 4-4. Lognormal Cumulative Density Functions for the Failure Data of Fan and Damper Given in Table 4-1

Table 4-2. Statistical Failure Data for Individual Components*		
Component	Failure to Operate or Run (1/hr)	Failure on Demand
Damper	Mean = 3.0×10^{-6}, Error Factor = 10	Mean = 3.0×10^{-2}, Error Factor = 10
Fan	Mean = 3.0×10^{-5}, Error Factor = 3	Mean = 5.0×10^{-3}, Error Factor = 5
*Blanton, C.H. and S.A. Eide. "Savannah River Site Generic Base Development." Rev 1. WSRC-TR-93-262. Aiken, South Carolina: WSRC. 1998.		

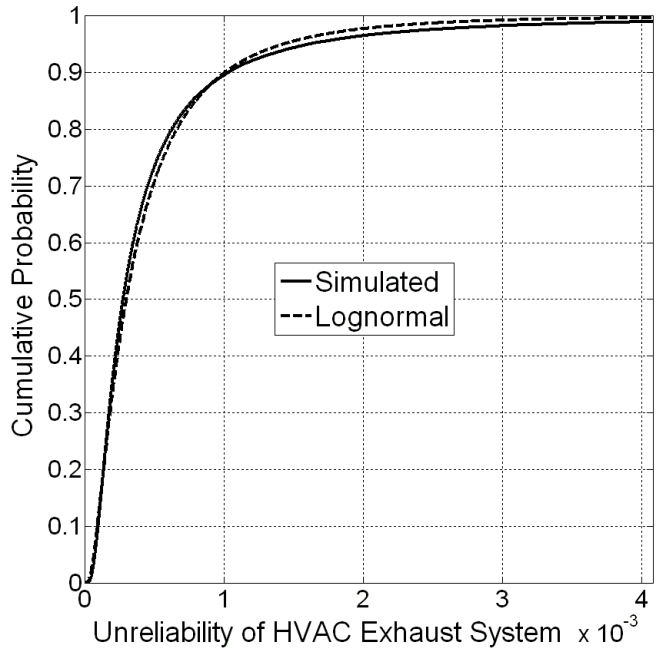


Figure 4-5. Distribution of the Unreliability of Heating, Ventilation, and Air-Conditioning Exhaust System

the top event probability will correspond to 2.0×10^{-3} and 4.0×10^{-3}, respectively. From these results, it can be concluded that the event sequence frequency is more sensitive to both the basic event of human operator failure and to the overall mission time, but less sensitive to the standby mission times (for the values discussed above).

Although not considered here, the generic failure data used as input could also be modified if a significant variation from other equally plausible sources of information indicates significant variation beyond the uncertainty ranges already covered in Table 4-2 (e.g., higher mean values and/or error factors). Previously mentioned common-cause failures of the system could also considerably dominate sensitivity results. As common-cause considerations are traditionally assumed to have the potential to directly affect the top event, assumptions about component dependency and control and operational procedures (e.g., inspection, testing, maintenance) could further affect the unreliability estimates described previously. This could also be an area to address though future preclosure safety assessment analysis.

4.4 Categorization of Event Sequences

The scenario of dropping a loaded cask onto a hard surface, described in Section 4.1.1, is modeled as an event tree, as discussed in Section 4.1.2, with considerations and input parameters discussed throughout Section 4.2. Using the initiating event frequency given in Section 4.2.1 and subsequent unreliability estimates of the heating, ventilation, and air-conditioning system to exhaust any potential releases, the event sequence frequencies can be estimated as shown in Figure 4-6. The heating, ventilation, and air-conditioning system unreliability corresponds to the component failure data shown in Section 4.2.2 (i.e., Table 4-2),

along with a heating, ventilation, and air-conditioning overall mission time of 168 hours, standby mission time of 24 hours, and human operator undependability of 10^{-2}. Assuming an operational period of 30 years in which transportation, aging, and disposal canister handling operations (including the scenario described in Section 4.1) would take place, the revised event

sequence frequency would be 3.3×10^{-6}/yr, resulting in a Category 2 event sequence. The categorization limit is established in 10 CFR Part 63 for event sequences that may occur before permanent closure of the potential geologic repository (i.e., event sequences with at least one chance in 10,000 of occurring before permanent closure). Applying this limiting value to the frequencies shown in Figure 4-6 indicates that Event Sequence B (initiating event followed by a successful heating, ventilation, and air-conditioning system operation) would be a Category 2 event sequence, while Event Sequence C is below the Category 2 limit. Both event sequences would be below the limits established for Category 1 event sequences (i.e., those expected to occur one or more times before permanent closure). For Event Sequence B, the expected number of drops would be 0.167 for a 30-year operational period, with 8.5×10^{-5} for Event Sequence C.

If the considerations of uncertainty and sensitivity analysis, presented in Section 4.3, are included, further scrutiny of the parameters affecting these event sequences may be warranted. For example, if the human operator undependability value of 10^{-1} is used, Event Sequence C would still not the reach Category 1 limit, but would become a Category 2 event sequence based on the mean, due to a modification in the probability P_2 (i.e., the unreliability of the heating, ventilation, and air conditioning system) as shown in Figure 4-6, with an event sequence frequency of 8.4×10^{-6}. Similarly, if the overall mission of the heating, ventilation, and air-conditioning system is increased to a 4-week period with a human error undependability of 10^{-2}, the resulting frequency would be 1.1×10^{-5}.

Considering the uncertainty analysis results in Section 4.3 in the Event Sequence C frequency, the resulting cumulative distributions assuming a 1-week, 4-week, or 8-week overall heating, ventilation, and air-conditioning mission time are presented in Figure 4-7. These results are obtained by running separate simulations for each overall heating, ventilation, and air-conditioning mission time. Along with the distributions, the Category 2 limit for a 30-year operational period and the corresponding mean values are also shown. Figure 4-7 reflects the change in categorization by increasing from a 1-week to a 4-week mission time. Furthermore, it also indicates the level of uncertainty that may result from the overall analysis of the event sequence frequency for categorization purposes.

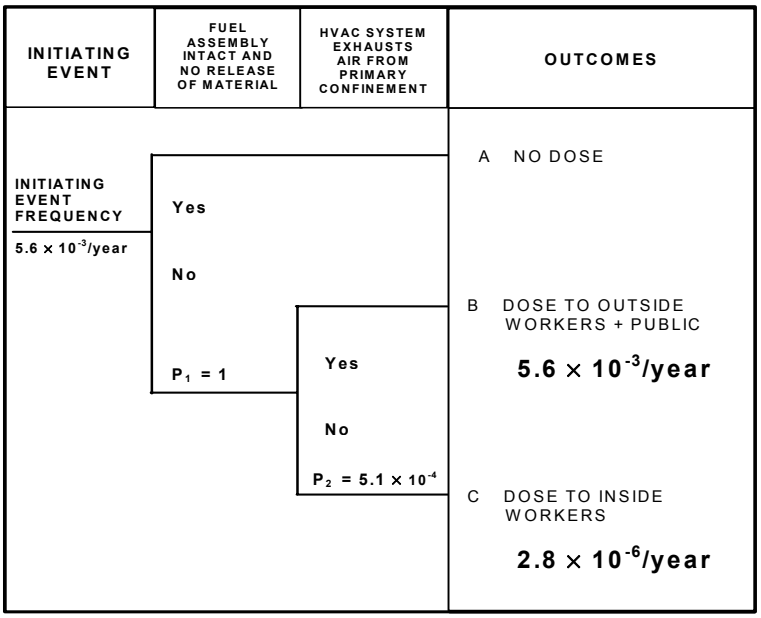


Figure 4-6. Event Sequence Frequency Estimation

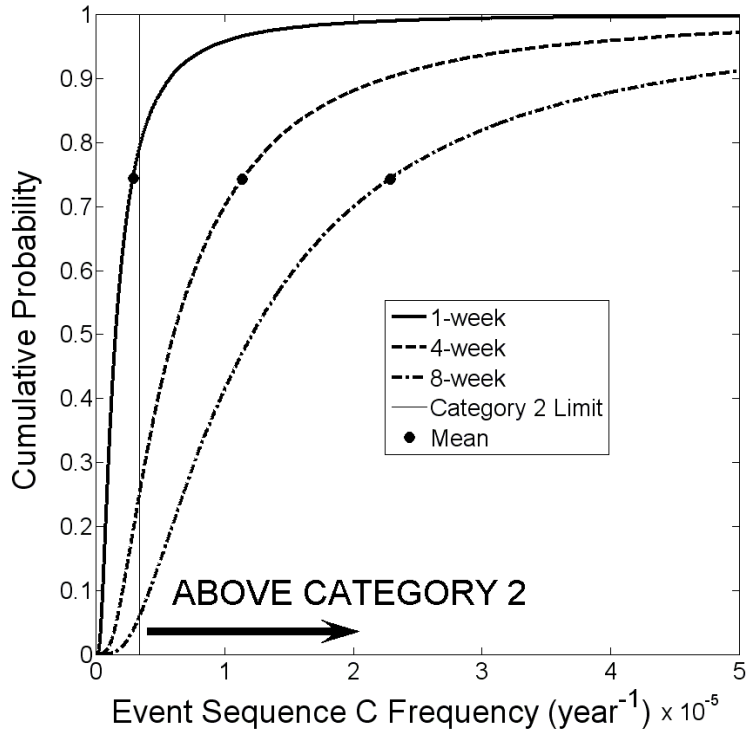


Figure 4-7. Uncertainty in Event Sequence C Frequency

5 CONSEQUENCE ANALYSIS FOR A CASK DROP

Data from 1968–2000 identify 11 events in which reactor fuel assemblies have been dropped during movement of the assemblies either prior to loading into a reactor, during loading, or after removal as spent fuel. None of the 11 dropped fuel assemblies failed (NRC, 2006). No radioactivity was released as a result of these events. The cited study empirically estimated the probability of fuel rod failure at a 6.4 percent level (NRC, 2006). The following parameters should be considered to more accurately estimate the number of damaged rods: the weight of the dropped load; the height of the drop; and the compression, torsion, and shear stresses of the fuel rods. The following consequence analysis does not include the probability of the cask drop event; therefore, all dose values are conditional on the occurrence of the drop event. Because the analysis does not account for the drop height, the fraction of potentially damaged fuel rods, or the contribution of direct radiation during normal operations, it should be considered as an example. Furthermore, because there were no recorded cases of fuel rod failures resulting from fuel assembly drops, all radiation dose estimates are normalized to one hypothetical failed spent fuel rod. Attention should be focused on the methodology as opposed to the numerical values of the normalized conditional dose. The dose output results are conditional on the drop event; therefore, the results are presented as conditional normalized doses.

5.1 Dose Calculations for Workers in Pool Room of a Conceptual Wet Handling Facility

5.1.1 Description of the Scenario

The scenario considered in this exercise assumes a drop of one shielded transfer cask loaded with a transportation, aging, and disposal canister during its transfer inside the conceptual Wet Handling Facility. A transportation, aging, and disposal canister contains 21 average (i.e., typical in terms of burnup, enrichment, and cooling time) pressurized water reactor commercial spent nuclear fuel assemblies. Commercial spent nuclear fuel characteristics of the average pressurized water reactor spent nuclear fuel are presented in Table 5-1. The shielded transfer cask is drained and contains air inside the cavity. A transportation, aging, and disposal canister is not drained and contains pool water inside its cavity. The shielded transfer cask lid is bolted; the transportation, aging, and disposal canister lid is not. Accidental drop of the shielded transfer cask takes place over a hard concrete floor during cask transfer. A fraction of the fuel rod cladding is breached and, as a result of the drop, radioactive materials are released into the pool room air. Workers present in the conceptual Wet Handling Facility pool room during cask transfer can potentially be exposed to these released radioactive materials and receive radiation exposure. The radiation dose to the potential recipients from the released gaseous radioactive materials is assessed for the scenario.

5.1.2 Methodology

5.1.2.1 Assumptions

Table 5-2 contains major parameters used in the dose assessment. The dose assessment does not include the probability of the event. The following assumptions are used in the dose assessment analyses.

Table 5-1. Characteristics of Average (Typical) Commercial Spent Nuclear Fuel Assemblies and Parameters, Burnups, and Cooling Times Used in Source-Term Derivation*	
Parameter Name, Units	**Parameter Value**
Assembly type	pressurized water reactor 15 × 15
Initial uranium enrichment, percent	4
Initial uranium loading, assembly, kg [lb]	475 [1,047]
Cobalt content, g/assembly [lb/assembly]	50 [0.11]
Burnup, GWd/MTU	48
Cooling time, years	25

*Bechtel SAIC Company, LLC. "Pressurized Water Reactor Source Term Generation and Evaluation." 00000C-MGRO-00100-000-00B. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2004.

Table 5-2. Parameters Used in Worker Pool Room Dose Calculations in Case of Accidental Drop of Shielded Transfer Cask on Hard Concrete Floor During Cask Transfer	
Parameter Name [units]	**Parameter Value**
Conceptual Wet Handling Facility pool room floor dimensions, m [ft]	23.5 wide × 62.5 long [206 × 77]*
Smaller mixing air volume, m^3 [ft^3] {Mixing air volume height: 2 m [6.6 ft]}	2.94×10^3 [1.06×10^5]*
Larger mixing air volume [m^3] {Mixing air volume height: 24.4 m [80 ft]}	3.59×10^4*
Gaseous radioactive materials airborne release fraction [-]	0.3†
Maximum time spent in mixing volume after release, min [s]	15 [900]‖
Spent nuclear fuel type [-]	Average pressurized water reactor‡
Inhalation rate, m^3/s	3.5×10^{-4} (default)¶

*Tooker, D. and T. Dunn. "Surface Facilities—Initial Handling Facility, Wet Handling Facility, and Receipt Facility, NRC/DOE Technical Exchange and Management Meeting on Design Changes Approved Through DOE's Critical Decision-1 (CD-1) Process." Bechtel SAIC Company, LLC. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2006.
†NRC. NUREG–1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities." Washington, DC: NRC. 2000b.
‡Bechtel SAIC Company, LLC. "Pressurized Water Reactor Source Term Generation and Evaluation." 000-00C-MGRO-00100-000-00B. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2004.
‖Nes, R. and R. Benke. "Potential Indoor Worker Exposure from Handling Area Leakage: Dose Calculation Methodology and Example Consequence Analysis." San Antonio, Texas: CNWRA. 2006.
¶Maxwell, T., B. Dasgupta, G. Adams, R. Benke, and N. Eisenberg. "Preclosure Safety Analysis (PCSA) Tool Version 3.0 User Guide." San Antonio, Texas: CNWRA. 2005.

1. The shielded transfer cask lid is assumed to be bolted, while the transportation, aging, and disposal canister lid is assumed to be unbolted, as discussed in Section 2.2.1. Furthermore, it is assumed that the transportation, aging, and disposal canister lid is not dislodged in the drop because of its heavy mass {i.e., approximately 300-mm [11.8-in]-thick lid}.

2. In accordance with NUREG/CR-6487 (NRC, 1996), gaseous species include H-3, I-129, and noble gases; 30 percent of gases contained in fuel rods are assumed to be released out of damaged fuel rods (NRC, 2003c; 2000a,b; 1997b; 1996) into the shielded transfer cask cavity. After water is drained out of the shielded transfer cask internal cavity, the cavity air temperature and pressure begin to rise. As a result, 30 percent of gaseous species contained in fuel rod gaps are released into the conceptual Wet Handling Facility pool room air. The assumed fraction of gases released from damaged rods is bounding; values reported in consensus standards for airborne release fractions are 4 to 60 times lower for noble gases (ANS, 1998). In addition, no credit is taken for retaining gases by bolted shielded transfer cask and unbolted transportation, aging, and disposal canister lids that are not likely to be dislodged.

   Therefore, it is expected that the actual airborne released fraction of gases would be lower than 30 percent. The released gases instantaneously mix with the air in the pool room, establishing a uniform concentration that is then reduced by ventilation. The entire amount of iodine released from damaged rods instantaneously leaves the shielded transfer cask. Retention of iodine by water in the transportation, aging, and disposal canister cavity will have to be accounted for as elemental and organic species separately, by applying so-called decontamination factors. If the depth of water above the damaged fuel inside the transportation, aging, and disposal canister is less than 7 m [23 ft], decontamination factors will have to be calculated on a case-by-case basis (NRC, 2003c, 2000a) using a method developed by Burley (1971). Therefore, the assumed amount of released iodine is bounding.

3. During shielded transfer cask transfer and in the course of the cask drop event, water inside the transportation, aging, and disposal canister remains below the boiling point; therefore, no bubble formation on the cladding surface would significantly affect the release of radioactive materials out of the shielded transfer cask.

4. Workers do not wear respiratory masks because the drop occurs in the course of normal operations. This assumption is consistent with nuclear industry procedures for fuel handling.

5. Dose calculations are made assuming a fully operational ventilation and air-conditioning system with an airflow rate of 40.12 m^3/s [85,000 cfm]. The system is discussed in Section 2.1.2.6.

6. A 15-minute average exposure time is assumed for workers present in the pool room (Nes and Benke, 2007). This value is considered bounding because the amount of released radioactive materials is expected to be sufficient to promptly activate the radiation alarm system in the room, and the workers would leave the area promptly after the alarm sound or supervisor command. This assumes an adequately reliable alarm, supervisory action, or worker judgment after a drop event has occurred.

7. The fuel rod cladding is assumed to be intact before the drop and the fission products available for release, in case of a breach of the cladding, are assumed to involve only those that had reached the gap and plenum region during utilization of the fuel in the power reactor. The mechanical shock of the drop event and a continuously increasing temperature and radial temperature gradient within the fuel rod during transfer and after the drop are assumed to be insufficient to cause additional release of fission products from the fuel matrix. This assumption is reasonable because the drop height is expected to be less than 0.3 m [1 ft].

8. The dose analysis assumes that all released Rn-220 instantaneously decays to Pb-212 after release into the pool room air due to its relatively short half-life of 56 seconds. This assumption is reasonable because no derived concentration guides for radon are available.

9. The intake of radioactive materials for the dose recipient depends on the assumed mixing air volume. Mixing air volume is determined by the assumed volume height. The calculations were performed for 2- and 24-m [6.56- and 80-ft] heights to span the range of mixing air volumes in the pool room for a recipient totally immersed in the cloud of radioactive materials. The 2-m [6.56-ft] height results in higher concentrations because the recipient is still totally immersed in the cloud of radioactive materials. The 24-m [80-ft] height is the least conservative. To produce more realistic air mixing volume values, factors such as released gas species densities; release duration; vertical temperature gradient in the pool room air; locations of heating, ventilation, and air conditioning system components inside the room; ventilation rate; and others should be considered.

5.1.2.2 Methods

The PCSA Tool Version 3.0.1 BetaC is used in the dose analyses (Dasgupta, 2002). The software changes implemented in Version 3.0 that led to the Version 3.0.1 BetaC were minor in nature and documented in Software Change Reports in compliance with the CNWRA Quality Assurance Program. The PCSA Tool is used to estimate the committed effective dose equivalent from inhalation, deep dose equivalent from submersion in released radioactive materials, and the skin dose received by exposed workers located in the conceptual Wet Handling Facility pool room. The total effective dose equivalent is calculated as the sum of the committed effective dose equivalent and deep dose equivalent (NRC, 2000b). Two sets of calculations are performed for (i) a smaller mixing volume corresponding to an air volume mass of 2 m [6.56 ft] in height and (ii) a larger air mixing volume corresponding to the entire pool room volume. A geometry factor accounting for the reduction of the amount of gamma radiation inside the finite cloud versus the infinite cloud was calculated using Eq. (3), where *GF* is the geometry factor (NRC, 1997b; Murphy and Campe, 1974).

$$GF = \frac{1173}{V^{0.338}} \tag{3}$$

where

V — mixing air volume (ft^3)

The resulting value for the smaller mixing volume is *GF* = 24 and for the larger mixing volume is *GF* = 10. A larger geometry factor leads to a smaller direct dose in comparison to an infinite cloud. Geometry factors do not apply to beta-emitting radionuclides, such as Kr-85, which contribute the most to the recipient skin dose.

5.1.3 Results

PCSA dose output results for smaller and larger air mixing volumes are presented in Figures 5-1 and 5-2. The skin dose and deep dose equivalent are predominantly caused by Kr-85. Radioisotopes H-3, I-129, and Pb-212, which is the daughter product of released Rn-220, are predominant in the committed effective dose equivalent contributing approximately 50, 25, and 25 percent, respectively.

5.2 Dose Calculations for Workers Outside the Conceptual Wet Handling Facility

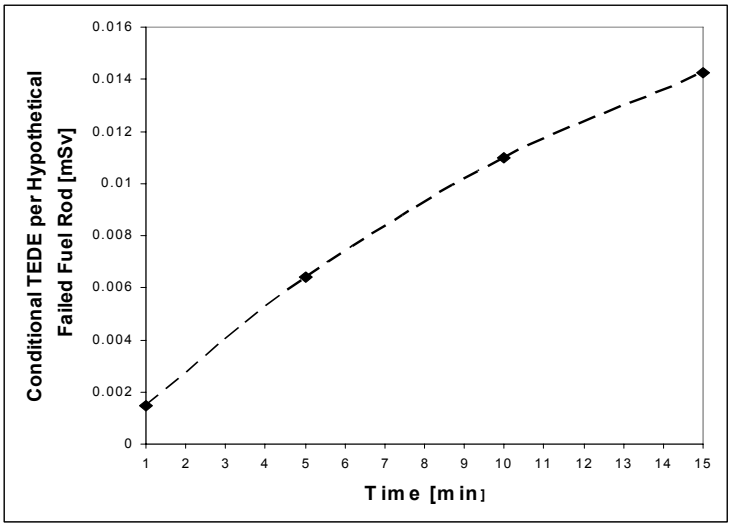
5.2.1 Description of the Scenario

This scenario considers a drop of one shielded transfer cask during its transfer inside the conceptual Wet Handling Facility described in Section 3.2. In this scenario, one shielded transfer cask is loaded with one transport, aging, and disposal canister. A transportation, aging, and disposal canister contains 21 average pressurized water reactor commercial spent nuclear fuel assemblies whose characteristics are presented in Table 5-2. The shielded transfer cask is drained and contains air inside the cavity. The transportation, aging, and disposal canister contains pool water inside its cavity. The shielded transfer cask lid is bolted; the transportation, aging, and disposal canister lid is not. An accidental drop of a shielded transfer cask takes place over a hard concrete floor during transfer. A fraction of the fuel rod cladding is breached as a result of the drop, and radioactive materials are released out of the shielded transfer cask into the conceptual Wet Handling Facility pool room air. The heating, ventilation, and air-conditioning system continues operating but its filtration function fails and radioactive materials are released outside the conceptual Wet Handling Facility through the ventilation exhaust. Workers present outside the facility can potentially be exposed to these radioactive materials and receive radiation exposure. Radiation dose caused only by these released radioactive materials is assessed for the potential recipients outside the conceptual Wet Handling Facility is conducted.

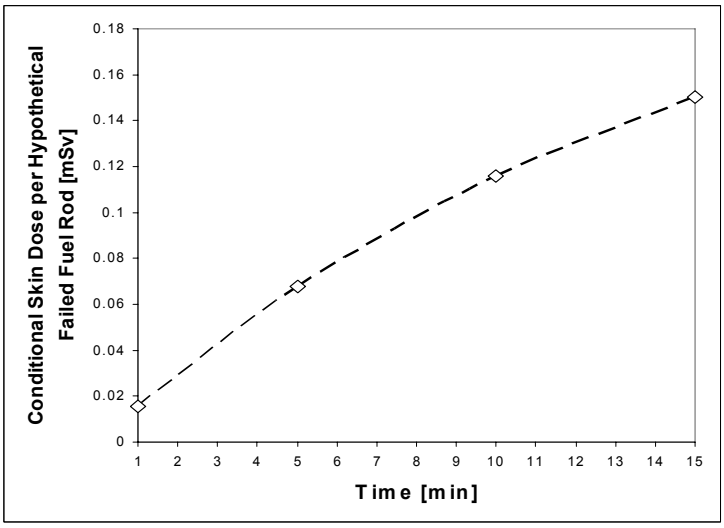
5.2.2 Methodology

5.2.2.1 Assumptions

Table 5-3 contains major parameters used in the dose assessment. The assessment does not include the probability of the event. Assumptions 1–5 and 8–9 listed in Section 5.1.2.1 are also valid for this dose assessment. In addition, the following assumptions are made specifically for the outside worker dose assessment analyses.

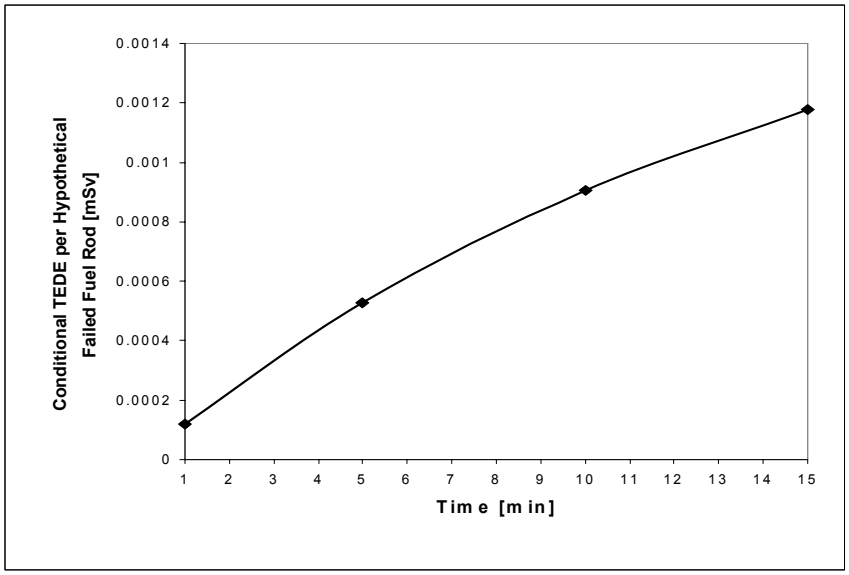


(a)

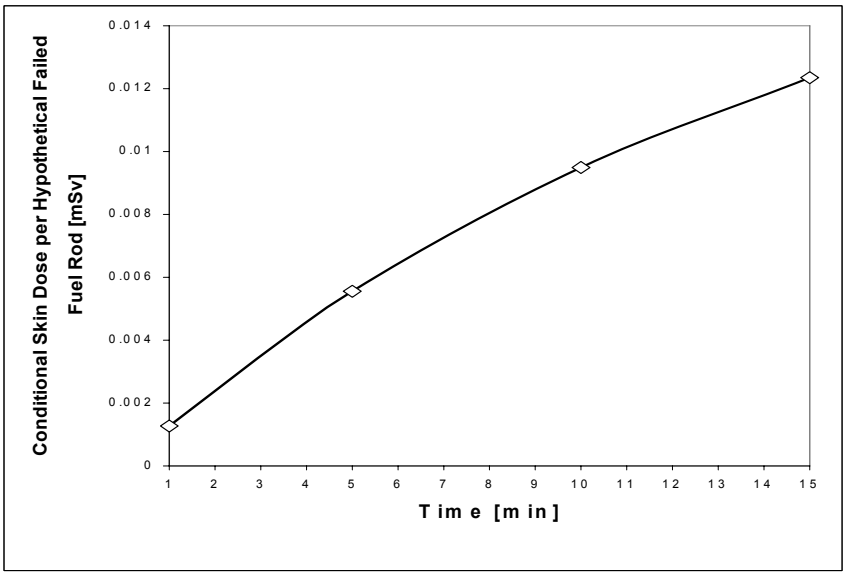


(b)

Figure 5-1. (a) Conditional Normalized Total Effective Dose Equivalent (TEDE) and (b) Conditional Normalized Skin Dose for Workers Inside the Conceptual Wet Handling Facility. If Site Transfer Cask Is Accidentally Dropped Workers Are Exposed to Radioactive Materials Released Into a Conceptual Wet Handling Facility Air Volume From the Cask. The Probability of the Drop Event Is Not Accounted For. Dose Values Are Normalized per Hypothetical Failed Fuel Rod for Relatively Smaller Mixing Volume {2-m [6.6-ft] Height Air Mixing Volume}.



(a)



(b)

Figure 5-2. (a) Conditional Normalized TEDE and (b) Conditional Normalized Skin Dose for Workers Inside a Conceptual Wet Handling Facility. If Site Transfer Cask Is Accidentally Dropped Workers Are Exposed to Radioactive Materials Released Into a Conceptual Wet Handling Facility Air From the Cask. The Probability of the Drop Event Is Not Accounted For. Dose Values Are Normalized per Hypothetical Failed Fuel Rod for Relatively Larger Mixing Volume {24.4-m [80-ft] Height Air Mixing Volume}.

Table 5-3. Parameters Used in Outside Worker Dose Calculations in Case of Accidental Drop of Shielded Transfer Cask Inside a Conceptual Wet Handling Facility Pool Room	
Parameter Name, Units	**Parameter Value**
Commercial spent nuclear fuel type	Average (typical) pressurized water reactor
Inhalation rate, m^3/s	3.5×10^{-4}*
Gaseous radioactive material airborne release fraction	0.3†
Conceptual Wet Handling Facility building side dimensions, width × height, m [ft]	80 × 24 m [262 × 79 ft]‡
Average wind velocity, m/s [ft/s]	1 [3.3]†§
Weather class	F †§

*Maxwell, T., B. Dasgupta, G. Adams, R. Benke, and N. Eisenberg. "Preclosure Safety Analysis (PCSA) Tool Version 3.0 User Guide." San Antonio, Texas: CNWRA. 2005.
†NRC. NUREG–1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities." Washington, DC: NRC. 2000.
‡Tooker, D. and T. Dunn. "Surface Facilities—Initial Handling Facility, Wet Handling Facility, and Receipt Facility, NRC/DOE Technical Exchange and Management Meeting on Design Changes Approved Through DOE's Critical Decision-1 (CD-1) Process." Bechtel SAIC Company, LLC. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2006.
§DOE. "Hazard Categorization And Accident Analysis Techniques For Compliance With DOE Order 5480.23. Nuclear Safety Analysis Reports." Standard DOE-STD-1027-92. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1992. CHANGE NOTICE NO.1 1997.

1. The heating, ventilation, and air-conditioning system continues at the normal operating level, but the filtration system is not credited for filtration of any released radionuclides. The system is discussed in Section 2.1.2.6. Using a method developed by Bechtel SAIC Company, LLC (2005), it is estimated that the conceptual Wet Handling Facility building pool room air contains 99 percent of the initially released radioactive material in approximately 44 minutes. Here, the ventilation rate is higher than that used in the indoor worker calculations. This relatively short exhaust of radioactive materials. (i.e., when compared to the typical worker shift duration of approximately 8 hours) creates a puff that passes over the receptor located 200 m [656 ft] outside away from the conceptual Wet Handling Facility building. This assumption is conservative because the outside worker is assumed to be exposed to approximately 100 percent of the initially released radioactive materials.

2. No credit is taken for plume depletion through radioactive decay of the relatively long-lived radionuclides before and after they are exhausted. This assumption is conservative because an actual plume would be depleted through radioactive decay.

3. The radioactive materials release heights are assumed to be 24 m and 0 m [79 ft and 0 ft] for one case considering and another not considering wake effects of the building. The building dimensions are discussed in Section 2.2.1.1.

5.2.2.2 Methods

RSAC Version 6.2 Code (INL, 2003) is used to estimate the committed effective dose equivalent from inhalation, deep dose equivalent from submersion, and the skin dose received by exposed workers located outside at a position 200 m [656 ft] away and downwind from the conceptual Wet Handling Facility building. Total effective dose equivalent is calculated as the sum of committed effective dose equivalent and deep dose equivalent (NRC, 2000b). The RSAC code uses a Gaussian plume model that ensures reasonable dose results only for receptors located at distances more than 100 m [328 ft] away. The code accounts for radioactive decay of released Rn-220 to Pb-212. Total effective dose equivalent is calculated by summing inhalation and submersion doses. The RSAC Version 6.2 code uses dose conversion factors from Federal Guidance Report 11 (EPA, 1988) and Federal Guidance Report 12 (EPA, 1993). Deterministic and probabilistic calculations for 100 realizations are performed with and without building wake effects. The following parameters were sampled in total effective dose equivalent probabilistic calculations:

- Average wind velocity
- Weather class
- Release fraction
- Released activity
- Deposition velocity
- Mixing layer height
- Breathing rate
- Decay time for exponential decay (credit for in-building decay)
- Activity mean aerodynamic diameter

Their probability distributions are available in Dasgupta (2002, Table 7-13).

5.2.3 Results

The results for conditional normalized total effective dose equivalent are presented in Table 5-4. The mean dose equivalent, the 5th percentile, and the 95th percentile dose equivalent values are conditional per cask event drop and normalized to one hypothetical failed fuel rod. The values are shown for cases with and without building wake effects. The dominant pathway in both cases is inhalation, contributing more than 99 percent of dose. The most dose-contributing radionuclides are H-3 and I-129 with 80 and 20 percent contributions, respectively.

Deterministic RSAC calculations assume constant stability class F (most probable, site-specific class) and a wind speed of 1.0 m/s [3.3 ft/s]. In probabilistic RSAC runs, stability class and wind speed are parameters sampled from distribution functions. The use of class F (moderate stability) leads to higher dose due to limited deviations of effluent concentrations. Compared to the case when stability class is sampled (allowing less stable atmospheric conditions to be considered), deterministic class F calculations estimate higher doses. In the building wake zone, contaminated and clean air mix, lowering the dose. This explains why doses considering building wake effects are lower, compared to doses that do not account for building wake effects. A constant building cross-sectional area adds significant corrections to the effluent concentration function that are not affected by statistical sampling. Under these circumstances, sampling stability class is less influential, explaining the difference between the spread of probabilistic estimates of the dose with and without wake effects. The number of 100 realizations adequately covers input parameter variation. Increasing the number of

realizations from 100 to 500 did not produce significant changes in total effective dose equivalent 5th and 95th percentiles or its mean value.

Table 5-4. Conditional Normalized Total Effective Dose Equivalent [mSv] for Outside Worker.

Percentile	Probabilistic Calculations		Deterministic Calculations	
	With Wake Effect	With No Wake Effect	With Wake Effect	With No Wake Effect
5th	6.15×10^{-7}	2.34×10^{-6}	8.84×10^{-6}	7.32×10^{-4}
Mean	4.77×10^{-6}	9.17×10^{-5}		
95th	1.06×10^{-5}	2.87×10^{-4}		

Notes: Dose values are conditional on a cask drop event that takes place inside a conceptual Wet Handling Facility. However, the probability of the event and the direct radiation dose for normal operations are not accounted for. Also, the worker is located outside at a position 200 m [656 ft] away from the conceptual Wet Handling Facility. Dose values are normalized per one hypothetical failed fuel rod. The RSAC Version 6.2 code was used in probabilistic and deterministic modes with and without consideration of building wake effect.

6 SUMMARY AND CONCLUSIONS

The information collected and discussed in Section 1 indicated that certain initiating events may be applicable to the conceptual spent fuel pool. These include (i) loss of cooling for pool water, (ii) loss of pool water, (iii) drops into or in the vicinity of the pool, and (iv) fuel assembly damage. Human error was the cause of some events (e.g., siphoning, which resulted in loss of pool water, and leaving interlocks inadvertently bypassed). Therefore, reviewers of a potential license application should be attentive to both the overall scenarios identified and, more specifically, areas where humans will be relied on to ensure safe fuel handling operations.

A single initiating event was analyzed in which a cask was dropped while being transferred from the spent fuel pool to the welding area of a conceptual Wet Handling Facility. The unmitigated drop was estimated to be a Category 2 event sequence. For a heating, ventilation, and air-conditioning system unreliability of 2.9×10^{-4}, the event sequence involving the drop followed by the heating, ventilation, and air-conditioning exhaust system continuing to operate, was also estimated to be a Category 2 event sequence. In this case, there was a potential dose to workers outside the building and to the public. Only conditional normalized dose values were presented (i.e., dose calculations did not incorporate the event sequence frequency). These dose calculations were part of a scoping analysis that may be useful to reviewers of a potential license application; as such, they should not be considered to be sufficiently accurate for decision making.

In addition to the dose calculations, an uncertainty and sensitivity analysis revealed that the scenario may need to be scrutinized more closely. For example, in terms of drop frequency alone, either 3 drops in 54,000 lifts (see Section 4.2.1) or 9 drops in 54,000 lifts (see Section 4.3) may be considered for the initiating event frequency. Although Category 2 in either case, this difference in number of drops from the same data source (NRC, 2003a), coupled with the uncertainty analysis (Section 4.3), may be important for reviewers to consider during the review of a potential license application.

7 FUTURE WORK

Some activities were originally considered but were not undertaken because of resource and time constraints. These activities are identified as areas of potential future work.

- Operations in the conceptual Wet Handling Facility to include: (i) cutting and welding operations, and (ii) receiving and shipping operations.
- Equipment in the conceptual Wet Handling Facility to include: (i) the spent fuel rack, and (ii) the spent fuel pool cooling system.
- Reliability analyses to include: (i) human reliability analyses and (ii) electrical power supply system reliability analyses.
- Identification of structures, systems, and components important to safety
- Development of risk insights.

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