Dry Cask Storage Probabilistic Risk Assessment Scoping Study

Technical Report

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1003011

Final Report, March 2002

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CITATIONS

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This report describes research sponsored by EPRI.

The report is a corporate document that should be cited in the literature in the following manner:

Dry Cask Storage Probabilistic Risk Assessment Scoping Study, EPRI, Palo Alto, CA: 2002. 1003011.

REPORT SUMMARY

This report describes and evaluates the current state of risk assessment methodologies applicable to dry cask storage probabilistic risk assessment (PRA) and suggests appropriate approaches for performing the various aspects of a dry cask storage PRA.

Background

The U.S. Nuclear Regulatory Commission (NRC) approved dry storage of spent fuel in Part 72, Title 10 of the U.S. Code of Federal Regulations (10CFR72). Since the 1980s, the number of dry casks stored at U.S. sites has grown substantially. However, dry storage licenses per 10CFR72 are limited to a term of 20 years from the date of issuance. This licensing period was selected because it was originally believed the federal government would be accepting spent fuel before these dry storage licenses expired. Since the federal government will not be accepting spent fuel until 2010 at the earliest, the industry is confronted with pursuing license renewals. This situation has led the industry and the U.S. NRC to identify methods for assessing the risks of the dry cask storage option.

Objective

To investigate and identify potential approaches for performing a probabilistic risk assessment for dry cask spent fuel storage.

Approach

The project team reviewed risk assessment methods applicable to dry cask storage and identified potential PRA approaches. The project's goal was to plan the development of a dry cask storage PRA, assuring it will address the most important safety issues.

Results

The report describes a dry cask storage PRA approach into appropriate supporting elements and investigates how the elements are best analyzed and integrated to provide PRA results and insights. This report does not document the development and results of a completed dry cask storage PRA; rather, it assesses applicable methodologies for developing such a risk assessment.

EPRI Perspective

Nuclear power plants were typically designed to store about 10 years of spent fuel. With the federal government's delay in accepting spent fuel from commercial plants, utilities have been considering other options for spent fuel storage. These options have included optimizing spent fuel pool storage space by re-racking spent fuel pools, rod consolidation, and removing non-fuel items. Such options offer finite expansion. As a means to create further expansion of interim spent fuel storage until the federal government can begin accepting spent fuel, utilities have turned to the use of dry storage technologies. This risk assessment will help provide risk insights for the regulatory process involving dry cask spent fuel storage.

Keywords
Dry cask
PRA
Spent fuel
Probabilistic risk assessment
Safety

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INTRODUCTION AND PURPOSE

Introduction

Since the inception of the U.S. commercial nuclear power industry, the U.S. federal government has maintained responsibility for the ultimate management and disposal of spent nuclear fuel. The Nuclear Waste Policy Act of 1982 mandated that the federal government was to begin accepting spent fuel from commercial utilities by 1998. For a variety of reasons the federal government has not met this schedule, and currently expects to begin accepting spent fuel by about 2010.

Nuclear power plants were typically designed to store no more than 1-2 decades of spent fuel. With the federal governments delay in accepting spent fuel, utilities have been pursuing other options for spent fuel storage. These options have included optimizing spent fuel pool storage space by: re-racking of spent fuel pools, rod consolidation, and/or removal of non-fuel items. Such options offer finite expansion. As a means to create further expansion of interim spent fuel storage until the federal government can begin accepting spent fuel, utilities have turned to the use of dry storage technologies.

Dry storage of spent fuel was approved by the U.S. NRC in Part 72, Title 10 of the U.S. Code of Federal Regulations (10CFR72). [61] The first plant to move spent fuel to dry storage was the Surry in the early 1980's. Since that time, the number of plants that have pursued this option and the number of dry casks stored at U.S. sites has grown substantially.

The dry storage licenses per 10CFR72 are currently limited to a term of 20 years from the date of issuance. This licensing period was selected as it was anticipated that the federal government would be accepting spent fuel before the dry storage licenses were to expire. As the federal government is not expecting to begin accepting spent fuel for about another decade, the industry is confronted with pursuing license renewals.

This situation has caused the industry and the U.S. NRC to pursue performance of risk assessment of the dry cask storage option. The U.S. NRC and the industry over the last three decades have performed risk assessments of the various phases of the nuclear fuel cycle:

- Burnup in the reactor
- Storage in the spent fuel pool
- Transportation offsite for permanent storage
- Storage at permanent repository

The U.S. NRC WASH-1400 study of 1975 was the first formal probabilistic risk assessment (PRA) of nuclear reactor operation. [83] In 1990 the U.S. NRC published an update to WASH-1400, the NUREG-1150 studies. [84] In the 1980's and 1990's the industry performed at-power risk assessments of both internal and external initiating events in response to the U.S. NRC Independent Plant Evaluations (IPE) and Independent Plant Evaluations of External Events (IPEEE) Program, as requested by Generic Letter 88-20. [85] In addition, the industry has been managing and assessing activities during shutdown.

The WASH-1400 study also investigated risk associated with spent fuel pool storage. Subsequent to WASH-1400, a number of additional risk assessments were performed to enhance the understanding of spent fuel pool risk, including the following:

- Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36, NUREG-0612, July 1980 [7]
- Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982, July 1987 [11]
- Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants, NUREG-1738, February 2001 [45]

Similarly, the U.S. NRC has performed a number of risk assessments concerning the transportation of spent fuel, including the following:

- Shipping Container Response to Severe Highway and Railway Accident Conditions, NUREG/CR-4829, February 1987 [48]
- Re-Examination of Spent Fuel Shipment Risk Estimates, NUREG-6672, March 2000 [58]

The risk assessment for permanent storage is the Yucca Mountain Repository Environmental Impact Study. [13]

Interim storage onsite in dry storage systems remains the (previously unanticipated) phase of the fuel life cycle that has yet to be investigated using probabilistic risk assessment. The projected continued use of dry storage for at least the next decade has made the performance of dry cask risk assessment a prudent undertaking by the U.S. NRC and the industry. Risk insights from such studies will be used to support risk-informed decisions, guide regulatory activities, and enhance public safety and confidence.

Purpose

The purpose of this project is to investigate and identify potential approaches for performing a probabilistic risk assessment for dry cask spent fuel storage. This risk assessment will be useful in providing risk insights to the regulatory process. The goal of providing this risk information is to assure that the regulations are formulated in a manner that addresses the most important safety issues while eliminating excess conservatisms and associated regulations in areas where safety is minimally impacted.

The purpose of this report is to describe and evaluate the current state of risk assessment methodologies applicable to dry cask storage PRA, and to suggest appropriate approaches for performing the various aspects of a dry cask storage PRA. The report divides a dry cask storage PRA into appropriate supporting elements and investigates how the elements are best analyzed and then integrated to provide PRA results and insights. This report does not document the development and the results of a completed dry cask storage PRA, but provides an assessment of the applicable methodologies for use in developing such a risk assessment.

2
PROJECT APPROACH

As discussed in the previous section, the purpose of this study is to review available information and to identify appropriate approaches for performing a dry cask probabilistic risk assessment (PRA). To that end, the following general approach was used in this project:

- Information Identification and Review
- Review U.S. NRC Dry Cask Storage Risk Assessment
- Identification of Potential PRA Approaches

Each of these project aspects is described below.

Information Identification and Review

This aspect of the project involved review of currently available information and analyses (e.g., U.S. NRC, EPRI, vendor, etc.). The purpose was to ensure that a sufficiently large base of existing information and knowledge is considered to support recommendation of appropriate dry cask PRA approaches.

Pertinent information was identified and obtained from the following sources:

- U.S. NRC public documents room (PDR)
- U.S. NRC ADAMS system
- DOE website
- U.S. National Laboratory websites
- Nuclear Energy Institute (NEI) website
- Organisation for Economic Co-Operation and Development (OECD) website
- Discussions with cask vendors [97]
- Discussions with U.S. NRC staff [96]
- NEI Dry Storage Conference May 2001 [98]

This information was reviewed and used to outline potential dry cask PRA approaches, as described in Section 4 of this report.

Refer to the Reference section of this report for the reference sources identified and used. Appendix A provides a bibliography of references, sorted by issue, potentially useful to analysts scoping and developing a dry cask PRA.

Review of U.S. NRC Dry Cask Storage Risk Assessment

A pilot PRA of a spent fuel dry cask storage system is currently being performed for the U.S. NRC Spent Fuel Project Office (SFPO) by the Office of Research (RES) to assess the potential risk to the public from the storage of spent nuclear fuel from a civilian nuclear reactor. [5,42,53] The purpose of this U.S. NRC project is to develop a method for performing a probabilistic risk assessment, for spent fuel dry cask storage systems, that will lead to recommendations as to whether it would be cost beneficial for PRAs to be performed for every cask design and site.

The U.S. NRC started this study in 1999. At the writing of this report, the U.S. NRC has not yet released this draft report. Based on publicly available documents and discussions with the U.S. NRC, the following provides an overview of the U.S. NRC study [96, 99]:

- Quantitative PRA using event trees and fault trees
- Accident sequences (i.e., event trees) fairly small, addressing the following top events: fuel integrity, cask integrity, building isolation, and recovery actions
- Specific cask analyzed: Holtec International HI-STORM 100 – a welded cask design
- Specific site analyzed: Edwin I. Hatch – a two unit General Electric boiling water reactor (BWR) in Georgia

Identification of Potential PRA Approaches

Based on the information gathering and review, this task outlines potential risk assessment approaches appropriate for producing a dry cask storage PRA.

A general overview of the developmental process of a dry cask storage PRA is provided in Section 3. Section 4 provides discussions of appropriate approaches for each of the key elements of a dry cask storage PRA.

3
OVERVIEW OF DRY CASK STORAGE PRA

Analysis Scope

The general scope of a dry cask storage probabilistic risk assessment (PRA) first needs to be defined. One characteristics that broadly aids in defining the scope of the analysis is the spent fuel life cycle (refer to Table 3-1). As can be seen from Table 3-1, accidents occurring during loading in the spent fuel pool, and accidents that occur during cask transportation to permanent storage at a repository are not in the scope of this dry cask storage risk assessment; such scenarios are addressed by other industry studies. [7,45,58,13]

Another aspect that defines the scope is the type of hazards (initiating events) to be assessed. PRAs typically categorize hazards into internal events (system or operator errors) and external events (natural phenomena, and similar hazards, external to the system). At-power reactor PRAs typically separate these types of hazards into discrete PRA studies. This report is written assuming that both internal and external hazards will be assessed and quantitatively incorporated into the dry cask storage PRA as appropriate.

Comparison of Dry Cask PRA and at-Power PRA

Due to the differences in the design, construction, and operation of a dry cask storage system compared to a nuclear power plant, there are significant differences among the key elements of a dry cask storage PRA and an at-power reactor PRA.

Initiating Event

- A dry cask storage system is designed around passive features, and does not include normally running operating systems and automatically initiated standby systems as does a nuclear power plant. As such, many of the internal initiating events that are appropriately modeled in at-power PRAs (e.g., turbine trip, loss of feedwater, loss of condenser vacuum, etc.) are not applicable to a dry cask system.
- Human errors, as well as equipment failures, are contributors to the frequencies of internal event initiators.
- The list of external hazards to be considered and dispositioned for a dry cask storage PRA and for an at-power PRA is the same.

Table 3-1
Dry Cask Storage PRA Scope

CASK LOADING/STORAGE PROCESS	DRY CASK STORAGE PRA SCOPE
Movement of empty cask into/through building	Outside Scope (1)
Lowering of empty cask into Spent Fuel Pool (SFP)	Outside Scope (1)
Loading of spent fuel into cask	Outside Scope (1)
Removal of loaded cask from SFP	In-scope
Closure of loaded cask: • Bolting or welding • Drying, inerting, testing, etc.	In-scope
Transfer of loaded cask through/out of building	In-scope
Transfer of multi-purpose canister (common feature of some designs) from transfer cask to storage cask (design dependent)	In-scope
Onsite transportation to Independent Spent Fuel Storage Installation (ISFSI)	In-scope
Storage at ISFSI	In-scope
Removal of fuel from storage cask and loading into transportation cask (design dependent)	Outside Scope (2)
Transportation from site and permanent storage at repository	Outside Scope (2)

NOTES:
(1) This portion of fuel cycle analyzed as part of U.S. NRC spent fuel pool risk assessment. [45]
(2) This portion of fuel cycle analyzed as part of U.S. NRC spent fuel transportation hazard [58] and Yucca Mountain [13] risk assessments.

Accident Sequence

- As a dry cask storage system does not incorporate water into the design, the inventory control critical safety function of an at-power PRA does not apply to a dry cask storage PRA. Similarly, the dry cask system is designed and loaded such that no active criticality control measures are required.
- The endstates in a PRA may be defined in a number of ways: fuel failure; containment failure; radionuclide release; dose; economic loss, etc.. At a high level, many of the end states options of a PRA are applicable to both an at-power reactor and a dry cask storage PRA. On a specific level, there are differences. For example, in the at-power PRA the core damage end state (often referred to as a Level 1 PRA) is not directly applicable to a dry cask PRA. However, the radionuclide release from the containment end state (often referred to as the Level 2 PRA) is generally appropriate. As another example, the potential dry cask PRA

end state of the frequency of Fuel Retrievability Failure is not applicable to an at-power PRA.

Systems Analysis

- Given the design and operation of a nuclear power plant, fault tree modeling of running and standby systems is a significant portion of an at-power PRA.
- Given the passive nature or dry cask storage systems, fault tree modeling of systems does not represent a significant portion of a dry cask storage PRA.

Human Error

- The typical human reliability analysis (HRA) concept of dividing human errors into the following categories is generally applicable to a dry cask storage PRA: pre-initiator errors, errors that result in an initiator; post-initiator errors, and recovery actions. These action types are briefly defined as follows:
  - Pre-Initiators: Pre-initiator human errors are latent errors, the effects of which, exist at the time of the initiating event. These errors are typically categorized into equipment miscalibration errors and equipment restoration errors following test or maintenance activities.
  - Errors Causing Initiating Events: Operator errors that directly cause an initiating event are typically included as a contributor in the initiating event frequency analysis of a PRA. An example of such an error is an operator lifting the incorrect electrical lead during a surveillance test which results in a spurious signal and causing a plant trip.
  - Post-Initiators: Post-initiator actions are performed in response to an initiating event. Such actions are typically proceduralized and included in operator training.
  - Recoveries: The term "recovery action" may apply to a wide spectrum of potential operator actions, but typically refers to actions to recover from or repair previously failed functions or equipment. Such actions may or may not be proceduralized.
- While an at-power PRA focuses significant attention on post-initiator operator errors, the significant portion of human error in a dry cask storage PRA is in the area of errors that produce initiating events.

Data

- The use of data to support initiating event frequencies and equipment failure rates (both independent and dependent) is applicable to both dry cask storage and at-power PRAs.
- However, given the passive nature of dry cask storage systems, the number of equipment failure probabilities estimated and included in a dry cask storage PRA is much less than an at-power PRA.
- As discussed earlier, at-power analysis began in the 1970s. Since then IPE and IPEEE have been performed on all U.S. reactors. In addition there are over 2000 operating years of experience with reactor operations. As such significant data has been accumulated over the

last 30 or more years. In comparison, dry casks have not obtained nearly as much operating experience nor have dry cask PRAs been performed, thus the amount data available is in comparison significantly limited.

Structural Evaluation

- Structural failures related to loss of coolant accidents (LOCAs) and other piping failures are not applicable to a dry cask storage PRA.
- Probabilistic assessment of containment structural failure in response to accident loads is applicable to both dry cask storage and at-power PRAs.
- While the containment structural assessment of an at-power PRA primarily focuses on containment failure probability due to quasi-static internal temperature and pressure increases due to loss of decay heat removal, a significant aspect of containment structural analysis in a dry cask storage PRA is on external loads (e.g., due to cask drop incidents).

Thermal Hydraulic Evaluation

- The use of a thermal hydraulic code such as MAAP has no application to a dry cask storage PRA.

Consequence Evaluation

- The radionuclide release estimates used in at-power PRAs considers an entire reactor core load of end-of-cycle fuel; whereas, the radionuclide estimates of a dry cask storage PRA is based on a certain number of fuel bundles (cask design dependent) that have been stored for some time (typically greater than or equal to five years).
- While the radionuclide release for a postulated reactor core damage accident must first proceed through building structures prior to reaching the environment, a dry cask storage accident may occur outside on the storage pad where no surrounding building structure typically exists.

Developmental Process of a Dry Cask Storage PRA

The development of a dry cask storage PRA follows the same key steps as a reactor PRA; differences (see above) between a dry cask storage PRA and reactor PRAs is in the details and levels of effort involved in the various supporting aspects of the PRA. Similar to a reactor PRA, once the purpose and scope of the analysis are defined, the development of a dry cask storage PRA can be broken down into the following steps:

- Information gathering and review
- Initiating event analysis
- Accident sequence development
- System modeling

- Human reliability analysis
- Database development
- Analysis of physical processes
- Radionuclide release
- Accident sequence model quantification
- Consequence analysis

As can be seen from the above steps, development of a dry cask PRA, like other PRAs, involves engineering knowledge and analytical expertise in a variety of different disciplines.

Although the following discussions are presented in a linear fashion, development of a PRA is not a strictly linear process, but in fact is iterative. For example, data analysis and human reliability analysis feed back into the initiating event analysis, system analysis, and the accident sequence analysis. Similarly, the analysis of physical processes and releases feed back into the accident sequence development portion of the PRA. Figure 3-1 provides a conceptual view of the process and the iterative nature of the development of a dry cask storage PRA that is anticipated.

Information gathering and review

The initial step in the development of a PRA is the identification and study of large amounts of pertinent information. The typical types of information to be assembled and used as input to the PRA are:

- Site specific plant, system, and site design
- Site specific and cask specific procedural and operational information including cask loading and transportation procedures and heavy load procedures
- Operating experience
- Natural phenomena data such earthquake and weather information
- Generic and specific equipment failure rates
- Existing accident analyses
- Appropriate PRA methods

Such information provides the basis for the development of all aspects of the PRA.

Initiating event analysis

The first major developmental step in the development of the PRA involves identification of initiating events that present challenges to the dry cask storage system design and associated loading and storage processes. This process can involve the identification of numerous discrete and very specific initiators. These are then typically grouped into initiating event classes based on similarities of challenge characteristics and system response.

Estimation of the initiating event frequencies will involve one or more of the following data inputs: operating experience information, human reliability analysis, component failure rates, external event hazards and other natural phenomena frequency information.

Accident sequence development

Following initiating event identification, the next step is the delineation of the accident sequence logic. This task involves the identification of the critical safety functions and delineation of the different possible combinations of successes and failures (i.e., of top events, which may address individual component, structural, or operator failures; systems; or functions) which lead to either successful or undesirable (e.g., radionuclide release) end states. End state categories are defined and applied to each of the accident sequences.

The accident sequence development is typically performed using event tree models. A separate event tree is typically developed for each initiating event category.

System modeling

The modeling of the event tree top events (typically referred to as system modeling) supports the accident sequence modeling. This portion of the analysis involves the identification of the appropriate success criteria and the associated failure modes (e.g., equipment failures or unavailabilities, operator errors, etc.) that lead to failure of the top event. This aspect of the PRA is typically performed using fault trees which use Boolean algebra operators to logically combine the various failure mode combinations.

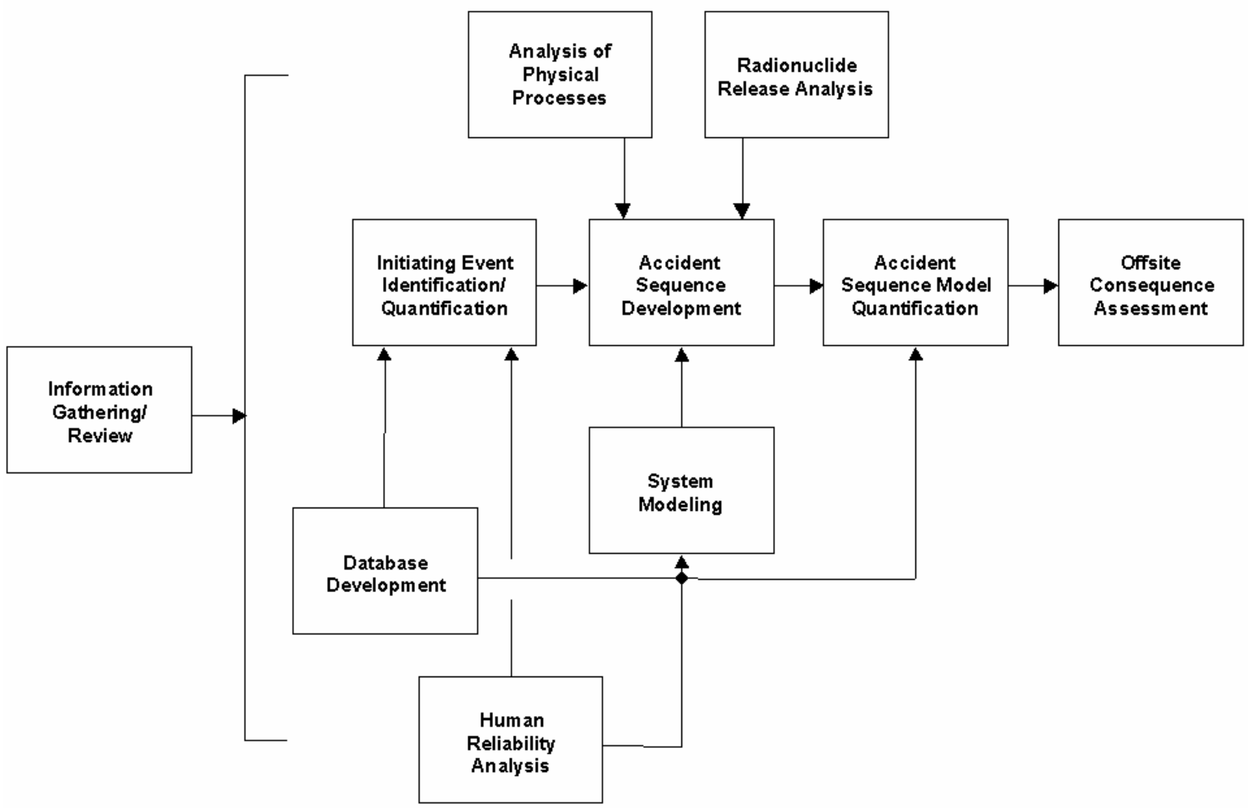

Figure 3-1
PRA Development Process

Human Reliability Analysis

Human reliability analysis (HRA) is a key task of a PRA. HRA both identifies the types of operator actions that play a role in initiating events and the accident sequence analysis, and quantifies the associated human error probabilities for the identified actions.

Database Development

Probabilistic accident sequence quantification ultimately requires the incorporation of probabilities for the various failure modes and events modeled in the PRA. This task involves the identification of appropriate generic and plant specific data and calculating the necessary failure probabilities. This information is maintained in a database and applied consistently to the failure modes included in the system and event tree models.

Analysis of Physical Processes

Analysis of the physical processes and phenomena associated with accident sequences is a key part of reactor PRAs. Computer codes specifically designed to model the physical processes associated with core damage accidents and their progression and mitigation are used in reactor PRAs to support accident sequence modeling, end state definitions, success criteria development, and operator action timing. This aspect of a PRA also applies to a dry cask storage PRA, but arguably to a lesser extent.

Radionuclide Release

Radionuclide release is a typical end state of a reactor PRA, and it is also an appropriate end state for a dry cask storage PRA. This step in the PRA development involves identification of the radionuclide inventory available for release for a particular cask storage accident and the assessment of the fraction of that inventory released from the cask for a given accident. This information is used to define representative release end states for the accident sequence analysis, and provides input for the consequence analysis (if a consequence analysis is performed as part of the PRA).

Accident Sequence Model Quantification

Accident sequence quantification is the integration of all the probabilistic and logical PRA model elements (i.e., initiating event frequencies, failure probabilities, human error probabilities, system models, and accident sequence models) and quantification of the PRA to produce accident sequence frequencies and related results (e.g., risk importance measures). The model quantification step involves checking for incorrect results, application of recovery factors (if appropriate), and identifying and correcting (as appropriate) cutsets containing multiple human errors. This step also may include the performance of sensitivity runs and uncertainty analyses.

The quantification step, like the other steps, is iterative and will often involve modifications of data and model structure given findings generated from initial quantification runs.

Consequence Analysis

The consequence portion of a PRA uses as input the radionuclide releases (source term) and frequencies from the accident sequence analysis to determine offsite consequences. The typical units of consequences are in units of frequency of early fatalities and latent fatalities. In addition to the use of source term information as input, this task assesses the impact of site-specific meteorological and population density to determine offsite consequences.

4 RECOMMENDED PRA APPROACHES

This section discusses appropriate approaches for performing the key elements of a dry cask storage PRA. The following PRA elements are discussed:

- Initiating Events
- Accident Scenarios
- Human Error Interface
- Systems Analysis
- Data Development
- Structural Evaluation
- Thermal Hydraulic Analysis
- Radionuclide Release and Consequence Evaluations
- PRA Computer Modeling and Quantification

Initiating Event Identification and Quantification

Initiating Event Identification

One of the most important aspects of risk assessment is the identification of potential hazards (i.e., initiating events). This aspect of the PRA involves the identification and discussion of the spectrum of potential challenges to a cask during onsite loading and transportation, and storage at the independent spent fuel storage installation (ISFSI). It also includes screening (or elimination) of challenges that can be dismissed with qualitative or conservative quantitative assessments.

Initiating events can be categorized into the following three major types:

- Operational events (e.g., cask drop)
- Natural phenomena (e.g., seismic event)
- Other external events (e.g., explosion)

The Yucca Mountain Environmental Impact Study and the spent fuel storage Standard Review Plans (SRP) (NUREG-1536 and NUREG-1567) provide a comprehensive list of challenges that should be considered (off-normal challenges, such as "drops less than design height," are not listed here as they are assumed to pose negligible challenges) [13,1,59]:

- Cask Tipover
- Cask Drop
- Flood
- Fire
- Explosion
- Lightning
- Earthquake
- Loss of Shielding
- Blockage of All Air Vents
- Tornadoes (and Associated Tornado Missiles)
- Nearby Facility Accidents
- Building Structural Failure onto Systems, Structures, and Components (SSC)
- Other Non-Specified Accidents

Certain external hazards are applicable or non-applicable given the site location being modeled with the dry cask PRA. For example, a tsunami would be applicable for consideration for a dry cask storage system located on the coast of the western United States.[65]

Other Non-Specified Accidents would include additional unique hazards identified during the analysis. This category would also include incorrect loading of the cask with newer fuel (which should be investigated as a potential accident initiator).

NUREG/CR-5042, Evaluation of External Hazards to Nuclear Power Plants in the U.S., provides a more detailed list of external hazards for consideration. [41] In addition, BMI/ONWI-551, Repository Pre-Closure Accident Scenarios, provides an extensive summary of potential operational accidents related to cask storage. [78]

Sabotage should be excluded from consideration as this challenge will be addressed by a separate more confidential study. The U.S. NRC is already involved in the performance of such an assessment. [33]

An additional item to note with respect to hazard identification is the difference between design basis events and beyond design basis events that are the focus of PRAs. Although the hazard type may be the same, the degree of the hazard may be different. For example, due to the use of single-failure-proof cranes, a particular cask may be designed to a design basis drop of 12 inches. This would not preclude the PRA from assessing the potential for multiple failures leading to a drop height in excess of the 12 inch design basis. Similarly, a particular cask may be designed for a specific design basis earthquake; again, this would not preclude the PRA from assessing the impact of seismic events greater than the design basis.

Once a comprehensive list of potential hazards is compiled, a formal screening process should be applied to identify those events that should be retained for further analysis and those which should be eliminated from any further evaluation. Such a process is desirable in that it would provide a traceable documented assessment of the hazards considered in the PRA. Such a screening process can be based on similar criteria as that used in the NUREG-1150 studies [52], i.e.:

1. The event is of equal or lesser damage potential for which the cask or ISFSI is designed
2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events.
3. The event cannot occur close enough to the site to impact a cask or the ISFSI.
4. The event is included in the definition of another event.
5. The event is slow in developing and there is sufficient time to eliminate the source of the threat or to provide an adequate response.

Similar screening techniques are used in DOE safety analysis studies (e.g., DOE/WIPP-95-2065, EGG-WM-10881). [68,40]

Initiating Event Frequency Estimation

The initiating events may occur due to one or more causes:

- Events during onsite loading and transportation
- Human errors
- Equipment failures
- Events during onsite storage
- Human errors
- Equipment failures
- Natural phenomena
- Onsite or offsite hazards (airplane crashes, runaway fork lifts)

Acceptable approaches to selected hazards are described below.

Cask Drop

A cask drop could be due to both equipment failures and/or human error. A fault tree approach may be used to quantify cask drop frequency. The fault tree would include crane and rigging equipment failure rates, and human error probabilities based on assessment of the cask loading and movement procedures (generic procedures are described in Chapter 8 of cask Safety Analysis Reports). Alternatively, crane failure rates are provided in a number of reference sources (including the U.S. NRC Spent Fuel Pool PRA) and could be used to estimate cask drop frequencies.

Aircraft Impact

The method for calculating aircraft hazard frequency is described in numerous risk assessments, industry studies (e.g., NUREG/CR-5042), and regulatory guidance documents (e.g., NUREG-0800, Section 3.5.1.6). [41,79] The following is a typical expression for calculating the annual probability of a random aircraft impact (for aircraft using Federal airways or aviation corridors) into a ground target [41]:

$$P_{FA} = C \times N \times A/w$$

where:

C = in-flight crash rate per mile for aircraft using airway

N = number of flights per year along the airway

w = width of airway in miles

A = effective area of target

Numerous sources of crash rate data for commercial operators are available for calculating the crash rate per mile (refer to Data Development Sub-Section). The number of flights on a particular air route can be obtained by contacting the pertinent regional Air Route Traffic Control Center or FAA Flight Service Station. The width of an airway can also be obtained from various sources (e.g., Federal Aviation Regulations for Pilots states that all federal airways are 8 nautical miles wide). NUREG-0800 Section 3.5.1.6 also provides guidance regarding the calculation of effective area (includes calculation of a skid area associated with a ground strike ahead of the target). [79]

U.S. Department of Energy (DOE) Standard 3014-96 provides a similar approach, and also provides detailed guidance (as well as examples calculations) for performing an aircraft impact frequency calculation. [74]

NUREG/CR-5042 and DOE-STD-3014-96 provide also provide expressions for calculating the frequency of civilian or military aircraft impacts. [41,74]

Natural Phenomena

Natural phenomena need not be considered an initiating event during the short time period of the cask life related to loading and onsite transportation. Definition and quantification of natural phenomena (e.g., seismic events, tornadoes) are dependent upon site location.

The frequency of seismic events and extreme winds (e.g., tornadoes, hurricanes) should use hazard frequency curves that account for hazard intensity as a function of occurrence frequency. Acceptable data sources for these and other external hazards are discussed in Data Development Sub-Section.

Frequency Units

Like reactor at-power PRAs, the frequency of initiating events for a cask PRA is in units of calendar year. However, unlike at-power PRAs, the calculation of initiating event frequencies for a cask PRA do not include adjustments for capacity factor (i.e., the likelihood the reactor is running when the event occurs).

Accident Sequence Development

This aspect of the PRA is one of the most important of the analysis because it formulates the end states to be examined and defines the accident scenarios that can result in the unacceptable end states. Included in this aspect is the definition of the critical safety functions to be met by the dry cask and the conditions that represent failure of the system to mitigate the initiating event challenges. This task also involves the definition of success criteria and the probabilistic logic development of the various branch points of the accident sequences.

Critical Safety Functions

The critical safety functions of a dry storage cask are in principle very similar to those of at-power PRAs

- Criticality control
- Pressure control
- Decay heat removal
- Containment

In practical terms, the first three critical safety functions will not be challenged during most of the quantified initiating events. Dry casks are designed and loaded such that criticality, pressure control, and decay heat removal challenges are well within the design capability of the system. Criticality, pressure control, and decay heat removal may be significantly challenged (and should be specifically assessed in the accident sequence analysis) during the following (or similar) potential hazards:

- Mis-loading of newer fuel
- Fire

Cask Containment

Disregarding the above initiating event, the critical safety functions to be specifically questioned in the accident sequence analysis of a dry cask PRA relate to containment. With respect to the cask itself, casks are typically designed with the following multiple barriers:

- Fuel cladding
- Inner cask containment vessel
- Cask outer pack (overpack or transportation cask)

Intact fuel rods comprise the majority of the contents of dry casks, and as such, the cladding on the fuel rods themselves prevent the release of gap fission products. However, cask designs also allow the loading of damage fuel rods and fuel debris as long as these are first loaded into canisters. Such canisters may be open to the environment (i.e., the canisters may be fitted with debris screens) inside the inner cask containment vessel. A cask PRA should make the assessment or assumption as to whether the analysis includes the possibility that the cask analyzed contains fuel with breached cladding (and whether or not this impacts accident sequence consequences).

The cask of a dry cask systems is either a welded or a bolted vessel. This inner cask provides the main confinement boundary of the spent fuel. It is designed to withstand operational loads, as well as accidents and extreme natural phenomena.

A cask outer pack, if part of the design, provides additional shielding as well additional protection for outside storage and/or transportation. Outer packs are typically designed of concrete and steel.

Building Containment

In addition to the cask itself, additional containment may be provided by a plant building if the accident sequence in question occurs while the cask is indoors. Such a building may be the reactor building, fuel handling building, or a separate cask maintenance/transfer building.

Unlike an at-power reactor severe accident PRA in which containment features such as heating, ventilation and air conditioning (HVAC) systems (e.g. standby gas treatment systems in BWRs) or passive features of a secondary containment typically receive little credit (due to the high temperatures and humidity of such severe accidents), HVAC systems may produce significant mitigation of any release in a cask accident sequence. Beyond ventilation system operation and capability, the mitigation credit of the building should also consider whether the building is closed or open at the time of the cask accident sequence.

Supplemental Containment Features

In addition to buildings, certain dry cask systems may include additional engineered features (e.g., earthen berms) around the ISFSI as supplemental shielding. The U.S. NRC considers such engineered features as important to safety (as discussed in Spent Fuel Project Office (SFPO)

Interim Staff Guidance (ISG) No. 14). [31] A cask PRA should appropriately address such supplemental features if they exist.

Post-Accident Recovery Actions

In general, post-accident recovery and mitigation actions would include:

- Visual and radiological assessment
- Installation of temporary shielding
- Repair of damaged cask (if possible)
- Cask unloading with movement of the contents to another cask or fuel pool

Specific procedures and equipment and equipment designed specifically for post-accident recovery and mitigation efforts may or may not exist for a given site. In any event, such actions are not expected to impact the consequence results for most accident sequences of a dry cask PRA (especially if the accident were occurring outside on the ISFSI). For accidents involving release of gaseous or volatile fission products from the cask inventory, much of the release of fission products would likely have occurred prior to effective implementation of any recovery actions. For loss of shielding accident sequences that do not include release of gaseous or volatile fission products, such recovery actions may impact the cask PRA if one of the consequence measures is worker exposure.

The cask PRA should qualitatively discuss or quantitatively analyze post-accident recoveries as appropriate to the systems and procedures assumed in the analysis.

Accident Sequence Development

After the appropriate critical safety functions for each initiating event are identified, the accident sequences can then be defined.

While event trees may or may not be used as the actual instrument for sequence quantification, event trees are always desirable in a PRA to graphically depict accident sequences and the resulting accident types. As in any risk assessment, event trees can be designed with numerous discrete top events that address individual phenomena, systems, and operator actions (large event tree modeling approach), or with a smaller number of functional top events (small event tree modeling approach).

In the most conservative approach, the accident sequence may simply be the beyond basis initiating event, which would be assumed to result in some percentage or all of the fission product inventory of the cask being released. The recommendations provided here are more reflective of the "small event tree" modeling approach.

Event structures may differ depending upon initiating event type and whether the event occurs inside a building or outside. For example, an initiating event that occurs inside a building would analyze the benefit of the building and appropriate HVAC systems in preventing or mitigating any offsite release, whereas an event that occurs outside on the ISFSI would not question such mitigation. An extreme natural phenomena initiating event that impacts a cask on the ISFSI may impact more than one cask, whereas an event that occurs inside a building during loading and transportation is likely to involve a single cask.

Likely top events to include in the accident sequence events trees for a cask PRA are:

- Initiating Event and Hazard
- Inner Cask Integrity
- Fuel Cladding Integrity
- Building Integrity
- Recovery and Mitigation

Structural or equipment damage to the surrounding building and/or plant from the initiating event would be outside the scope of a dry cask PRA (other than how it directly relates to mitigation of the cask release). Such risk contributions are addressed in the industry in separate analyses (Heavy Loads PRA, and Spent Fuel Pool PRA). [7,11,45]

Binary or greater optional outcomes may be assumed for one or more of these top events. For example, the cask integrity can be modeled with “intact,” “leak,” or “failed” (ruptured) outcomes. Similarly, the fuel cladding integrity may be modeled with Intact (0% rods failed), “minor” (1% rods failed), “moderate” (10% rods failed), and “gross” (100% rods failed) failure outcomes. An example event tree structure for a drop or similar initiating event that occurs inside a building is shown in Figure 4-1.

Accident Sequence Endstates

The endstates of accident sequences in a cask PRA may be defined in various ways and units, such as:

- Failure of cask containment
- Failure of retrievability of fuel
- Release of fission products from cask
- Dose to onsite workers
- Dose at Site Boundary
- Economic Cost

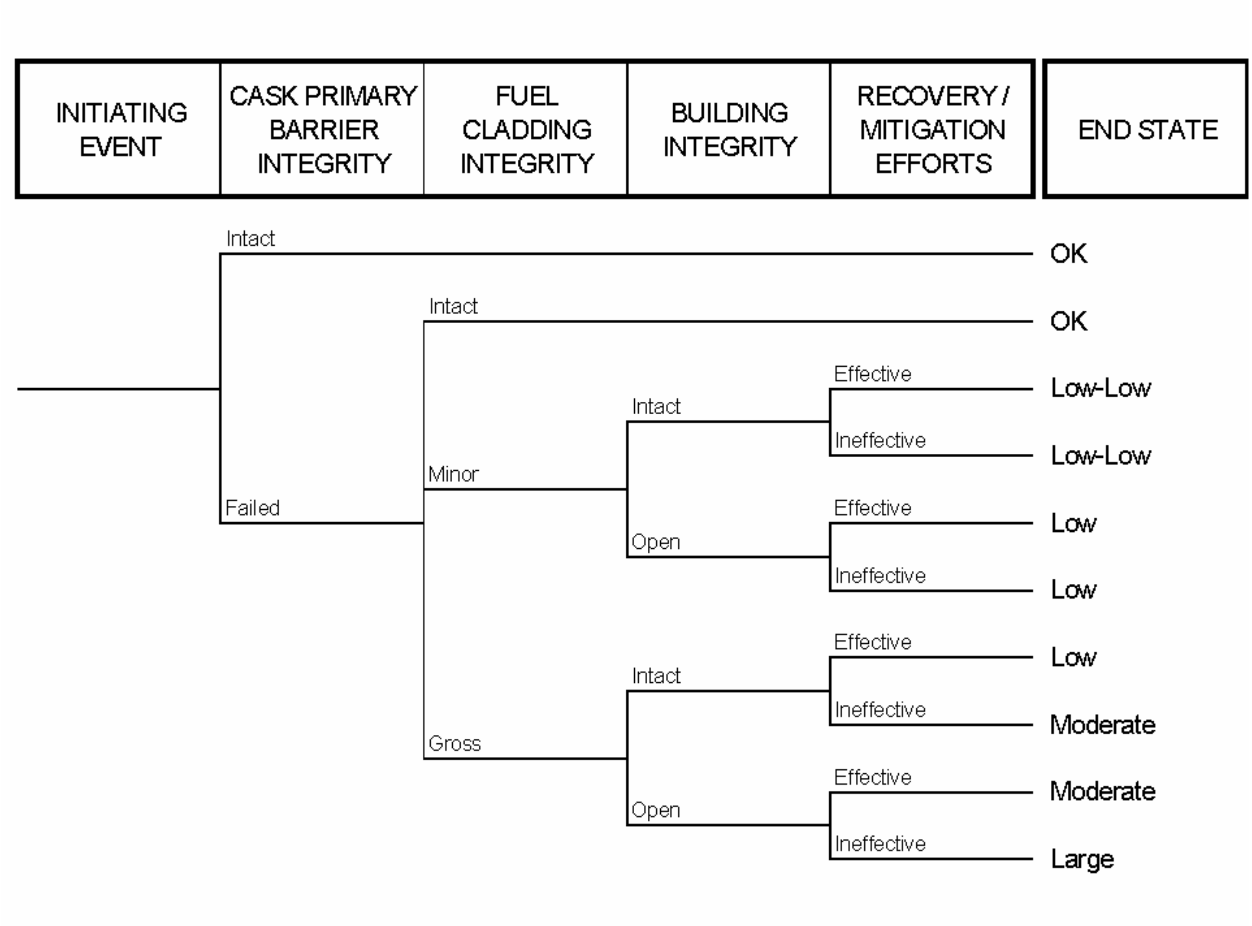

Figure 4-1
Example Event Tree Structure

Certainly a dry cask PRA may employ one or more such endstates, recognizing that some of the above are more difficult to assess and less broadly applicable than others. However, to be as consistent as possible with approaches used in U.S. industry at-power PRAs and to maximize the applicability of the base end state definitions, it is recommended that the endstates most appropriate for a dry cask storage PRA would be fission product release frequency and/or dose at the site boundary (or similar surrogate).

Dry cask systems are designed and maintained so that the spent fuel is retrievable such that it will not pose operational safety problems during removal from storage. Industry guidelines exist with respect to what constitutes degraded spent fuel.[66] In addition, industry examinations of dry stored spent fuel have been performed to investigate potential spent fuel degradation during long-term storage.[44] However, the use of fuel non-retrievability as an endstate should not be one of the base end state of a dry cask PRA.

As is typical in at-power PRAs, it may be preferable to define a discrete set of endstate categories (or bins) that cover the spectrum of potential consequences. Such a categorization would be reflective of dose calculations performed. An example list of endstates is as follows:

- OK
- Low-Low
- Low
- Moderate
- Large

The category names listed above may be renamed to avoid confusion with endstate categories used in at-power PRAs, and to reflect the fact that the consequences of postulated dry cask accidents are much less than that of postulated at-power reactor accidents (i.e., the radionuclide inventory and the energy in the spent fuel in a cask is much less than that in an operating reactor – refer to the Radionuclide Release and Consequence Evaluation Sub-Section for further discussion on this issue).

Human Error Interface

Human errors may occur during fuel loading into cask, cask decontamination and closure, transportation inside building, transportation onsite to storage pad, and during storage on the pad. Human errors may initiate an accident or contribute to the initiation of an accident, can contribute to failure of mitigation activities, or even exacerbate the accident. Some examples include cask drop during movement, improper sealing, and misalignment of support or monitoring systems.

Identification and quantification of human errors should be supported by review of cask operating experience events, performance of standard Human Reliability Analysis (HRA) techniques, and observation of cask loading and transportation operations.

Operating Experience

Operating experience event summaries related to dry storage casks can be found in a variety of sources:

- U.S. NRC Information Notices, Bulletins, Inspection Reports and related documents and correspondence
- Industry studies and reports

U.S. NRC Information

The following are selected U.S. NRC documents that provide discussions of operating experience related to dry cask storage:

- Bulletin 96-04: Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks [18]

- U.S. NRC Briefing on Status of Dry Cask Storage Issues, Public Meeting, Thursday May, 30, 1996. [19]
- IN 97-51: Problems Experienced With Loading and Unloading Spent Nuclear Fuel Storage and Transportation Casks [17]

In addition to the above, various U.S. NRC Inspection Reports related to dry cask storage systems and their operation provide discussions and background of operating experience.

Industry Studies

Sparse information in the form of industry studies of human error and operating experience related to dry cask storage systems currently exists. One such study has been performed but the content and conclusions of that study are generally qualitative in nature and directed toward transportation risk. That study, A Review of the Effects of Human Error on The Risk Involved in Spent Fuel Transportation, was performed for the Nebraska Energy Office in 1986 (and revised in 1987). [64] However, review of this study provides a background that may be useful in scoping out the human factor influences to be addressed in a dry cask storage PRA.

Human Reliability Analysis

General procedures related to the loading, onsite transportation, and onsite storage of dry casks are contained in the various cask Safety Analysis Reports (SAR) and may be used to facilitate the identification of human errors and the calculation of the associated human error probabilities.

It is recognized that, due to the nature of dry cask storage systems and associated postulated accident scenarios, that formal human reliability analysis may not play as significant and explicit a role in dry cask PRA as in at-power PRAs. In any event, various human error probability methodologies exist and these should be applied in the dry cask PRA where appropriate. The EPRI cause-based HRA methodology (as documented in EPRI TR-100259 [80]) is used in commercial nuclear power PRAs and would be an appropriate methodology, supplemented by manipulation error probability information from NUREG/CR-1278 (Handbook of HRA). [81] The U.S. NRC Accident Sequence Evaluation Program HRA Methodology (NUREG/CR-4772) may also be used; this methodology is used in safety analysis reports of various INEL facilities, as well as commercial nuclear power PRAs. [82]

If possible, the HRA should be supported by walkdowns, review of applicable cask procedures, and interviews of operations personnel to identify appropriate performance shaping factors (PSFs). Such information gathering affords an enhanced ability to identify the potential for human errors in the loading and storage processes and to properly quantify the associated human error probabilities. The HRA analysis should also account for dependence among operator action error probabilities, where appropriate.

Systems Analysis

Systems analysis is used here as a generic term to describe the development of logic models for the top events of the accident sequence models. Such models are best developed using fault trees. Refer to References [3] and [104], as necessary, for details regarding fault tree modeling techniques.

Recommended modeling features to incorporate, as appropriate, include:

- Component level failure modes
- Independent and dependent failure events
- Human error probabilities
- Developed support system logic

If the dry cask PRA is being performed for a specific site, the at-power PRA models for the site would likely contain logic models for support systems. Such existing fault tree logic may be used directly, with any necessary modifications, in the dry cask PRA. Fault tree logic for alternating current and direct current systems should be readily available; other systems that may be modeled in the dry cask storage PRA, such as hoists and building ventilation systems, are likely not available from the at-power PRA.

Data Development

Data development covers the identification and use of appropriate failure rate and event occurrence data to support estimation of initiating event frequencies and equipment failure rates. Human error probabilities are discussed separately in the Human Error Interface Sub-Section.

Initiating Event Frequency Data

Initiating event frequency estimation has been discussed previously in Initiating Event Identification and Quantification Sub-Section. As can be seen from that discussion, data inputs that may be required to calculate initiating event frequencies include:

- Crane failure rates
- Aircraft crash rates
- Onsite vehicle crash rates
- Natural phenomena occurrence rates
- Other external hazards

Crane Failure Rates

Crane failure rates are used to estimate the frequency of a cask drop accident. Crane failure can be due to crane equipment failure, rigging failure, and/or human error. A number of sources of pertinent failure data are available:

- WASH-1400 crane failure rate (3E-6 events per hour) [83]
- NUREG/CR-4982 [11] and EPRI NP-3365 (3E-6 events per hour crane mechanical failure; 3E-6 events per hour crane electrical failure; and 6E-4 events per lift human error) [57]
- U.S. Navy operating experience

The WASH-1400 3E-6 events per hour crane failure rate is an assumed value. EPRI NP-3365 re-examined the issue and determined that 3E-6 events per hour was too low of an estimate for crane failure, as listed above. The NUREG/CR-4982 study, performed three years later, used the same crane failure rates as EPRI NP-3365. The U.S. NRC's 1980 heavy loads risk assessment, NUREG-0612, used 1974-1977 crane failure information. The NUREG-0612 study includes a fault tree assessment for calculating load drop frequency.

The U.S. NRC re-assessed the NUREG-0612 heavy load risk in their recent Spent Fuel Pool Accident Risk Report. [45] The study included a re-calculation of the NUREG-0612 cask drop initiating event fault tree. This recent study by the U.S. NRC considered the following additional sources of information regarding crane failure:

- 1996-1990 U.S. Navy crane operating experience
- WIPP/WID-96-2196, "Waste Isolation Pilot Plant Truddock Crane System Analysis" [67]
- NEI data on spent fuel pool (SFP) cask lifts

Appendix 2C of the recent U.S. NRC spent fuel pool risk assessment provides a background discussions on the data, as well as calculational details of cask drop frequency. [45]

The Yucca Mountain Environmental Impact Study also uses U.S. Navy crane operating experience to calculate cask drop frequency (Attachment X to MGR Design Basis Events Calculation). [13]

Aircraft Crash Rates

Numerous sources of crash rate data for commercial operators are available for calculating the crash rate per mile:

- The NUREG-0800, Section 3.5.1.6 recommends a value of 4E-10 events per mile for airways with less than 100 flights per day. [79]
- NUREG/CR-5042 cites a value of 2.298E-8 events per mile for U.S. certificated route, supplemental, and commercial operators of large aircraft. [41]
- The NUREG/CR-4550 Peach Bottom Study cites a value of 2.76E-9 events per mile (95th percentile) for commercial aircraft. [2]

- The Limerick Severe Accident Risk Assessment cites a value of 4.5E-10 per mile for U.S. commercial aviation for the years 1970-1975. [86]
- U.S. Department of Energy Standard 3014-96 provides crash rates for numerous types of aircraft. [74]

The U.S. NRC uses the DOE data in their most recent Spent Fuel Pool risk assessment [45]; as such, DOE-STD-3014-96 may be considered the preferential reference.

Onsite Vehicle Crash Rates

Sparse references sources exist for collision and accident rates of onsite industrial vehicles (e.g., forklifts). However, if such a hazard survives the initiating event screening analysis, quantification of the initiating event frequency of such an event can be based on analyses and accident rates used in DOE safety analysis reports (e.g., DOE/WIPP-95-2065, EGG-WM-10881). [68,40] Various potentially applicable failure rates are included in these DOE studies, such as:

- Frequency of forklift collision with waste packages due to hardware failure (e.g., brake failure, stuck accelerator): 2.6E-6 events per operation-hour.
- Frequency of forklift collision with waste packages due to human error: 5.0E-6 events per operation.

The above accident rates may be specific to the DOE facilities and operations. Assuming that human error is the dominant contributor to this hazard, human reliability analysis techniques (e.g., operator manipulation error rates from NUREG/CR-1278 [81]) may be used as an alternative approach to calculating the frequency of this hazard.

Natural Phenomena Occurrence Rates

The following selected phenomena are discussed here:

- Seismic Events
- Extreme Winds
- External Floods
- Lightning
- Forest Fires

Seismic Events: Seismic hazard curves are typically available in site Individual Plant Examination of External Event (IPEEE) studies. Alternatively, seismic hazard frequencies for U.S. reactor sites are available both from NUREG reports and EPRI reports. EPRI seismic hazard curves produced by EPRI are documented in EPRI NP-6395-D (April 1989) and EPRI NP-4726 (November 1988). [87,88] The U.S. NRC seismic hazard curves (calculated by Lawrence Livermore National Laboratory) are documented in NUREG-1488. [50] The U.S.

NRC hazard curves are generally more conservative in frequency. Use of the EPRI seismic hazard curves would be appropriate for use in an EPRI dry cask storage PRA.

Extreme Winds: The frequency of extreme winds (e.g., high winds, tornadoes, hurricanes), if determined to be a probabilistically significant hazard, may be based on site-specific information obtained from the site final safety analysis report (FSAR) or IPEEE submittal; otherwise, it may be based on industry studies or information from the National Severe Storms Forecast Center (or similar agency).

External Floods: The frequency of external floods, if determined to be a probabilistically significant hazard, may be based on site-specific information obtained from the site FSAR or IPEEE submittal; otherwise, it may be based on industry studies or information from the Army Corps of Engineers (or similar agency).

Lightning and Forest Fires: The frequency of lightning strikes or forest fires, if determined to be a probabilistically significant hazard, may be based on based on industry studies (e.g., DOE safety analysis reports) or information from the Bureau of Land Management (or similar agency).

Other External Natural Phenomena: Most external natural phenomena will most likely be screened out in the initiating event identification phase of the analysis (i.e., the phenomena is within the design of the cask, encompassed by another analyzed hazard, or of very low frequency, etc.). If frequency information for such phenomena is required, useful guidance may be obtained from NUREG/CR-5042 [41] and DOE safety analysis reports (e.g., EGG-WM-10881 [40]).

Other External Hazards

As discussed above for other external natural phenomena, other miscellaneous external hazards will most likely be screened out in the initiating event identification phase of the analysis. Such hazards would include external explosions, accidental missile strikes, etc. Given the wide variety of such miscellaneous hazards, pertinent specific reference sources are not provided here, rather a search of industry studies should be performed to identify appropriate data reference sources. For example, a risk assessment of accidents involving cruise missile impacts was performed in support of the safety analysis report for the Private Fuel Storage Facility. [34]

Equipment Failure Rates

Random Failure Rates

The characterization of equipment failures using either generic database or operating experience data may be required to support the quantification of certain specific initiating events and/or the quantification of accident sequences. For example, building ventilation equipment will most likely be included in certain accident sequences. As such, failure rates for fans, dampers, electrical equipment, etc. would be necessary to quantify the failure rate for the ventilation systems.

Equipment failure rates should be available from the at-power PRA for a given site, and this may be considered the most desirable source (if not simply for consistency sake). Alternatively, generic equipment failure probabilities can be found in the EPRI advanced light water reactor (ALWR) database (as documented in the EPRI ALWR Requirements Document, TR-016780), as well as other PRA industry sources (e.g., NUREG-1150). [89,84]

Dependent Failure Rates

Common cause failure (CCF) probabilities should be calculated using either the Alpha Method or the Multiple Greek Letter (MGL) method using the latest industry common cause failure information, as documented in NUREG/CR-5497. [90]

Structural Evaluation

This aspect of the PRA involves the assessment of structural fragilities related to integrity of the cask. Such assessments may include buildings (e.g., Cask Transfer Facility) in addition to the cask itself.

The purpose is to support initiating event definition as well as the accident sequence analysis. For example, if a particular cask design basis related to tornadoes covers the complete spectrum of tornado intensities (i.e., meteorological theory and experience suggest that tornadoes cannot exceed a certain wind speed), then this fact would likely be sufficient to preclude explicit quantitative analysis of tornado induced cask accident sequences. Conversely, if a particular cask design basis related to drop accidents covers only a certain spectrum of drop heights, and the PRA analyst determines that greater drop heights are not necessarily precluded from a PRA consideration, then a structural analysis of the cask would be useful to support the cask containment failure probability given a certain drop height.

Given the complexity of the cask systems, in terms of various load configurations and structural design considerations, finite-elements codes (e.g., ANSYS) are often used in the design basis structural calculations of casks. If possible to obtain and manipulate these models, it would be useful to perform beyond design basis loading calculations for accidents to be included in the dry cask PRA. However, such deterministic calculations also will require engineering interpretation to arrive at containment failure probabilities given certain loading conditions. Various civil engineering firms were employed by nuclear utilities in the 1980’s and 1990’s in support of the IPE program to perform such deterministic calculations and associated failure probabilities assessments for primary containment structures.

Alternatively, structural analysis information contained in the specific cask system safety analysis report may be used to calculate structural fragilities. Structural failure probabilities may be calculated with the available design basis information as input, using median capacity and fragility concepts set forth in past risk assessment studies (e.g., NUREG/CR-2300, “PRA Procedures Guide”; EPRI NSAC-60, “Oconee PRA”). [3,91] If available from cask design basis information, the median structural capacity for a given load may be calculated using the strength factor (yield stress/design stress) and the allowable ductility ratio. The structural failure probability would be calculated using a normal distribution calculation approach:

$$\text{Structural Failure Probability} = \Phi\left(\ln(\text{load/median capacity})/\beta_c\right)$$

where, β_c is the combined randomness and uncertainty inherent in materials performance and engineering analysis/knowledge.

Alternatively, more simplistic and conservative assumptions can be made in the dry cask PRA in lieu of structural analyses. For example, the more conservative approach would be to assume (i.e., probability equal1.0) complete containment failure for any accident loading beyond the design basis. A less conservative approach would be to assign a range containment failure probabilities (conservatively estimated based on judgment).

Thermal Hydraulic Analysis

Thermal hydraulic analyses typically provide a significant supporting role in the development of at-power PRAs. The EPRI MAAP code is a common software tool for performance of thermal hydraulic analyses to support PRA development. [92] Such analyses are used in at-power PRAs primarily to determine systemic and functional success criteria, and accident sequence timings.

In the case of a dry cask storage PRA, the use of a thermal hydraulic code such as MAAP has no obvious application. The MAAP code (and other related thermal hydraulic codes used in support of PRAs) have been designed specifically for operating reactors and their containment structures. One obvious dissimilarity is that a dry cask storage system does not employ water as a coolant. Another is that the fuel in a dry cask is spent and maintained in a geometric condition designed to preclude criticality. One possible application may be to investigate timings associated with a mis-loading of new fuel accident scenario during the short time window when the cask is undergoing the loading and closure process.

However, thermal analyses may play a role in a dry cask PRA depending upon the initiating event hazard screening process. For example, a dry cask is designed for a particular design basis fire exterior to the cask. Such thermal heat-up calculations are performed using a code such as ANSYS (which is capable of both steady state and transient calculations). If a beyond design basis fire is maintained for treatment in the dry cask PRA then it may be useful to obtain and manipulate (if possible) the design basis models.

Radionuclide Release and Consequence Evaluation

This aspect of the PRA involves the definition and assessment of the radionuclide releases associated with the modeled accident sequences. The calculated consequences should be used as the basis for the definitions of the accident sequence end states (refer to the Accident Sequence Development Sub-Section above). The following discussions assume that the end states to be used in the dry cask PRA relate to radionuclide release and/or dose at some distance.

Industry studies have been performed to calculated spent fuel consequences (for both spent fuel risk and for dry storage). Those that may be useful in estimating consequences for a dry cask PRA include:

- Best Estimate Offsite Dose from Dry Storage Cask Leakage, SMSAB-00-03 [93]
- Containment Analysis for Type B Packages Used to Transport Various Contents, NUREG/CR-6487 [62]
- Direct Exposure From Degraded Spent Nuclear Fuel Storage, Tetra Tech NUS [60]

Other potentially useful sources for such generic information include References [13], [57], and [70].

Consequence studies of spent fuel pool consequences are not listed here. Although such studies may be useful in a general sense with respect to understanding methodologies, the consequences of spent fuel pool accidents are typically not directly applicable to a dry cask PRA. The spent fuel pool accident consequences are based on fuel failure occurring in the spent fuel pool and typically involve a much larger (e.g., recent core offload) and fresher (e.g., less than 1 year) inventory of spent fuel than dry cask accidents.

One major difficulty in integrating or comparing dry cask storage risk with at-power risk is that there are no well understood surrogate risk measures such as exist for at-power operation (i.e., core damage frequency (CDF) and large, early release frequency (LERF)). The end states and corresponding success criteria are substantially different than those used in the at-power situation and at-power PRAs. The situation is similar to that of spent fuel pool risk. The U.S. NRC has stated the following in NUREG-1353 [8] which indicates LERF is not feasible for spent fuel pool risk:

> *"It is difficult to compare the estimated release frequency due to a spent fuel pool accident to a target value of $1x10^{-6}$ per reactor year for a large release. The spent fuel pool source term is not similar to the core damage (or melt) source term and the consequences of a spent fuel pool accident are dominated by latent cancer risks. A definition of a "large release" currently being considered by the staff is a release that has a potential for causing an offsite early fatality. This definition, with consideration for early fatalities, appear to suggest that the spent fuel pool release is not a "large release."*

The statement above is generally applicable as well to dry cask storage risk. As the estimated consequences will be used to support dry cask PRA end state category definitions, the end state definitions should be explicitly differentiated, as appropriate, from similar at-power PRA end states.

PRA Computer Modeling and Quantification

A quantitative PRA involves the development of computer models, and the integration and quantification of the models to produce accident sequence frequencies and other risk measures (e.g., importance measures). Assuming the modeling and quantification of the dry cask PRA will be performed using the EPRI CAFTA suite of PRA software codes, appropriate PRA modeling and quantification processes are as follows:

- Event trees designed in the ETA code and then the accident sequence logic exported to CAFTA fault tree logic.
- Individual system and nodal fault trees designed in CAFTA.
- CAFTA accident sequence logic files and system and nodal fault tree files merged into a single CAFTA file using the MERGER utility.
- PRAQuant code used to quantify the individual accident sequences.
- Necessary mutually exclusive, recovery, and flag files developed and the names and paths included in the PRAQuant file.
- A single reliability database developed using the Reliability Database Editor to hold the probabilistic and descriptive information for the initiating events, phenomena events, component failure events, and other parameters used in the PRA. The reference source for each parameter cited in the reference fields of the database.
- Sensitivities performed by either manipulating cutsets using the Cutset Editor, or by performing model modifications and rerunning sequences in PRAQuant.

5 CONCLUSIONS

With the increase in interim dry storage of spent nuclear fuel at reactor sites, both the U.S. NRC and the nuclear industry have considered appropriate the performance of risk assessments of the dry cask storage option. Other than an early 1980's comparative risk assessment of spent fuel storage options, EPRI NP-3365 [57], no dry cask storage PRA has yet been completed in the industry. The U.S. NRC, at the writing of this report, is planning and developing a risk assessment of a welded dry cask storage system at a specific U.S. reactor site.

This report investigates and identifies appropriate approaches for development by the industry of dry cask storage risk assessments. It is acknowledge here, that without a complete dry cask storage PRA having yet been completed, the recommendations in this report may be enhanced or modified once one or more such studies are completed and lessons-learned realized.

Although differences in details and levels of effort in certain analytical areas differ between reactor PRAs and a dry cask storage PRA, the developmental process is the same. The key developmental elements include:

- Initiating Events
- Accident Sequence Development
- Systems Analysis
- Human Reliability Analysis
- Data Development
- Structural and Deterministic Analyses
- Radionuclide Release and Consequence Evaluations
- Accident Sequence Quantification

This report provides discussions of general approaches to address these elements and suggests useful reference sources for each element.

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This appendix provides industry references potentially useful to analysts scoping and developing a dry cask PRA. These references are categorized by the following general areas:

- General Background (Table A-1)
- Initiating Events and Operating Experience (Table A-2)
- Accident Sequence Development (Table A-3)
- Human Reliability Analysis (Table A-4)
- Structural (Table A-5)
- Consequences (Table A-6)
- Regulations and Regulatory Guides (Table A-7)

In addition to the above references, the various dry cask system Safety Analysis Reports provide useful and detailed information on the following:

- System Design Bases and Descriptions
- Structural Evaluations
- Thermal Evaluations
- Shielding Evaluations
- Criticality Evaluations
- Operating Procedures
- Accident Analyses

Such SARs (e.g., References [73,76,77], as well as others) can be found and downloaded from the U.S. NRC ADAMS document system.

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B
ACRONYMS

Table B-1 provides a list of selected acronyms related to the nuclear power industry, probabilistic risk assessment, and dry cask storage. A more comprehensive list of acronyms can be found in the U.S. NRC document NUREG-0544, NRC Collection of Abbreviations.

Table B-1
List of Selected Acronyms

ADAMS	[U.S. NRC] Agency-wide Documents Access and Management System
ADS	Automatic Depressurization System
ALARA	As Low As Reasonably Achievable
AO	Air-Operated
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
BHEP	Basic Human Error Probability
BOC	Break Outside Containment
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CET	Containment Event Tree
CHR	Containment Heat Removal
COC	Certificate of Compliance
CST	Condensate Storage Tank
CV	Check Valve
CW	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCS	Dry Cask Storage
DCSS	Dry Cask Storage System
DHR	Decay Heat Removal
DOE	U.S. Department of Energy
ECCS	Emergency Core Cooling System
ECOM	Error Of Commission

Table B-1 (continued)
List of Selected Acronyms

EDG	Emergency Diesel Generator
EIS	Environmental Impact Statement
EOM	Error of Omission
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guidelines
ESF	Engineered Safety Feature
FP	Fire Protection System
FSAR	Final Safety Analysis Report
FV	Fussell-Vesely
FW	Feedwater
HCLPF	High Confidence of Low Probability Of Failure
HEP	Human Error Probability
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation And Control
IE	Initiating Event
IORV	Inadvertently Opened Relief Valve
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance
ISLOCA	Interfacing Systems Loss of Coolant Accident
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LOSP	Loss of Offsite Power

Table B-1 (continued)
List of Selected Acronyms

MAAP	Modular Accident Analysis Program (software code)
MGL	Multiple Greek Letter (CCF methodology)
MGR	Monitored Geologic Repository
MMI	Modified Mercalli Intensity
MOV	Motor Operated Valve
MPC	Multi-Purpose Canister
MSIV	Main Steam Isolation Valve
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
OBE	Operating Basis Earthquake
PCS	Power Conversion System
PDS	Plant Damage State
PGA	Peak Ground Acceleration
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSF	Performance Shaping Factor
PSHA	Probabilistic Seismic Hazard Analysis
PSSA	Probabilistic Shutdown Safety Assessment
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAW	Risk Achievement Worth
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RISC	Risk Informed Safety Classification
RLE	Review Level Earthquake

Table B-1 (continued)
List of Selected Acronyms

RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth
RWST	Refueling Water Storage Tank
SAR	Safety Analysis Report
SBGT	Standby Gas Treatment System
SBO	Station Blackout
SFP	Spent Fuel Pool
SFPO	Spent Fuel Project Office
SGTR	Steam Generator Tube Rupture
SMA	Seismic Margin Assessment
SNF	Spent Nuclear Fuel
SORV	Stuck Open Relief Valve
SP	Suppression Pool
SPC	Suppression Pool Cooling
SPSA	Seismic Probabilistic Safety Assessment
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSC	Systems, Structures And Components
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSI	Soil Structure Interaction
SW	Service Water
TAF	Top of Active Fuel
THERP	Technique For Human Error Rate Prediction (See NUREG/CR-1278)
TSC	Technical Support Center
UHS	Uniform Hazard Response Spectrum

C
TERMS AND DEFINITIONS

The following is a list of selected terms (and associated definitions) related to the nuclear power industry, probabilistic risk assessment, and dry cask storage. The following terms and definitions are taken from a variety of industry sources, including: NUREG-1536, 10CFR72, and the ASME “Standard For Probabilistic Risk Assessment For Nuclear Power Plant Applications.”

***Best Estimate*:** The point estimate of a parameter utilized in a computation which is not biased by conservatism or optimism. Generally, the mean value of a parameter is considered to be the best estimate.

***Common Cause Failure*:** A common cause failure is a single, shared event that adversely affects two or more components at the same time. When the consequences of the event include the occurrence of an accident sequence initiating event, the event is called a common cause initiating event.

Confinement: Per 10CFR72, those systems, including ventilation, that act as barriers between areas containing radioactive substances and the environment.

Cutsets: Minimum combination of a set of events that, if they occur, will result in an undesired event such as the failure of a system or the failure of a function.

Core Damage: Uncovery and heatup of the reactor core to the point where prolonged oxidation and severe fuel damage is anticipated.

Damaged Spent Fuel: The classification of spent fuel as damaged or undamaged is a necessary element in the design, loading, and storage of dry cask storage systems. Cask licenses permit spent fuel with normal minor damage that does not present any safety issues. Spent fuel assemblies that do not exceed the permitted damage level are termed “undamaged,” while those assemblies that do exceed the permitted damage level are termed “damaged.” Protocols for determining the level of spent fuel damage are provided in U.S. NRC Spent Fuel Project Office Interim Staff Guidance ISG-1, “Damaged Fuel,” and NEI’s “Fuel Classification Protocol for Dry Fuel Storage and Transportation.” [66]

Deterministic: Differences in the paradigms, traditions, and functions of various organizations have resulted in separate approaches to risk assessment - often referred to as deterministic and probabilistic approaches. A deterministic analysis is one in which the inputs and conditions are assumed to be constant and accurately specified. The U.S. NRC defines deterministic approaches as those that "consider a set of challenges to safety and determine how those challenges should be mitigated." This approach assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a pre-determined set of design basis events. It then requires that the licensed facility design include safety systems capable of preventing and/or mitigating the consequences of those design basis events to protect the public health and safety.

End State: An end state is the set of conditions at the end of an event sequence that characterizes the impact of the sequence on the plant or the environment. In most at-power reactor PRAs, end states typically include: success states, plant damage states for Level 1 sequences, and release categories for Level 2 sequences. Major Level 2 end state groups and subgroups identify groups of release categories with similar potential for offsite consequences.

Event Tree: An event tree is a quantifiable logical network that begins with an initiating event or condition and progresses through a series of branches (usually binary, but may be three or more branches at a given node) that represent expected system or operator performance that either succeeds or fails and arrives at either a success or failed condition (e.g., core damage) at the end of the tree.

Event Tree Top Event: Top events are the conditions that are considered at each branch point of an event tree. They may address system behavior or operability, human actions, or phenomenological events. A particular event tree sequence can be described in terms of the status of the plant relative to each top event.

External Event: An external event is an event that initiates outside of the plant systems that can affect the operability of plant systems. An earthquake or a missile generated by a tornado are examples of external events, as well as fires within the plant.

Failure Rate: Equipment failure rates can be demand dependent or time dependent. The failure rate of a component is the conditional probability of failure on the next demand (for a standby component) or in the next hour of operation (for an operating component), given it has not already failed.

Fault Tree: A fault tree is a logic diagram that is used to determine the logical combination of failure or condition causes that will produce an undesired event. Fault trees are generally used to determine and quantify the logical combination of causes that would result in failure or unavailability of a system modeled in the event tree models.

Fault Tree Top Event: A fault tree top event is the event at the very top of the fault tree, sometimes referred to as the undesired event, for which the fault tree determines the causes.

Figure of Merit: Figure of Merit is the quantitative value obtained from a PRA analysis used for evaluating the results of an application. For at-power PRAs, typically these include core damage frequency (CDF) and large, early release frequency (LERF).

Frequency: Frequency is the number of occurrences of an event per trial or per unit time.

Fussell-Vesely Importance (F-V): Fussell-Vesely Importance of a modeled plant feature (usually a component, train, or system) is defined as the fractional decrease in total risk level (usually CDF) when the plant feature is assumed perfectly reliable (failure rate = 0.0). If all the sequences comprising the total risk level (e.g. CDF) are minimal, the F-V also equals the fractional contribution to the total risk level of all sequences containing the (failed) feature of interest. Note that F-V = 1-1/RRW. (See Risk Reduction Worth.)

Hazard: Hazard is a source of danger or consequence.

Human Reliability Analysis: Human Reliability Analysis is the quantitative evaluation of human performance considered in PRAs. Human Error Probabilities (HEPs) are the resulting quantified parameters of HRA, and are human failure rates (as contrasted with equipment failure rates defined earlier).

Independent Spent Fuel Storage Installation (ISFSI): A facility designed, constructed, and licensed per 10CFR72 for the purpose of interim storage of spent nuclear fuel.

Initiating Event: An initiating event is any event that perturbs the steady state operation of the plant, if operating, or the steady state operation of the decay heat removal systems during shutdown operations such that a transient is initiated in the plant. Initiating events trigger sequences of events that challenge the plant control and safety systems.

Internal Event: An initiating event originating as the result of plant system failures or human errors (as contrasted with external initiating events defined earlier). A turbine trip or a LOCA are examples of internal event initiators.

Large, Early Release Frequency (LERF): A large, early release, as used in at-power PRAs, is a radioactivity release from the containment which is both large and early. Large is defined as involving the rapid, unscrubbed release of airborne fission products to the environment. Early is defined as occurring before the effective implementation of the off-site emergency response and protective actions.

Mission Time: The mission time is the time that a system or component is required to operate in order to successfully perform its function.

Performance Shaping Factor (PSF): Performance Shaping Factors (PSF) are those factors which influence human error rates. Typical PSFs include level of training, quality and availability of procedural guidance, time factors, etc.

Plant Damage State: Plant damage states are collections of accident sequence end states according to plant conditions at the onset of severe core damage. The plant conditions considered are those that determine the capability of the containment to cope with a severe core damage accident. The plant damage states represent the interface between the Level 1 and Level 2 analyses.

Probabilistic: Contrasted to a deterministic analysis, a probabilistic analysis is one in which the inputs and conditions are assumed to be variable. The U.S. NRC defines probabilistic approaches as those that (1) [Allow] consideration of a broader set of potential challenges to safety, (2) [provide] a logical means for prioritizing these challenges based on risk significance, and (3) [allow] consideration of a broader set of resources to defend against these challenges.

***Probabilistic Risk Assessment* (*PRA*):** PRA is a quantitative assessment of the risk associated with plant operation and maintenance. The risk is measured in terms of the frequency of occurrence of different events, including severe core damage. In general, the scope of a PRA is divided into three categories: Level 1, Level 2, and Level 3. The Level 1 scope maps from initiating events to plant damage states including their aggregate, severe core damage. Level 2 includes Level 1 mapping from initiating events to release categories (source term). Finally, Level 3 includes Level 2 and uses the source term of Level 2 to quantify consequences, the most common of which are health effects and property damage in terms of costs. Of course, the scope of a PRA may vary considerably within each Level depending on the preferences of the plant owners or the regulators.

***Probabilistic Safety Assessment* (*PSA*):** Alternative term for probabilistic risk assessment.

Probability: Probability is a numerical measure of a state of knowledge, a degree of belief, or a state of confidence about the outcome of an event.

Recovery Factor: A correction factor (i.e., a failure probability in the range of 0.0 to 1.0) applied to cutsets or accident sequences to account for the possibility that the modeled accident sequence will be ultimately corrected prior to reaching the modeled end state. A recovery factor typically models operator actions to recover previously failed equipment.

Release Category: Release categories are typically the end states of the Level 2 portion of a PRA. Release categories characterize major classes of accident sequences in terms of the nature, timing, and magnitude of the release of radioactive material from the plant during a severe core damage accident. The factors addressed in the definition of the release categories include the response of the containment structure, timing, and mode of containment failure; timing, magnitude, and mix of any releases of radioactive material; thermal energy of release; and key factors affecting deposition and filtration of radionuclides.

Retrievability: Interim spent fuel storage systems must be designed per 10CFR72 to allow ready retrieval of spent fuel for further processing or final disposal. Per NUREG-1536, ready retrievability is the capability to return the stored radioactive material to a safe condition without the release of radioactive materials to the environment or radiation exposures in excess of the limits defined by 10CFR20. Per 10CFR72(h)(1), spent fuel cladding must be protected during storage against degradation that leads to gross ruptures of the fuel or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

Risk: Risk encompasses what can happen (scenario), its likelihood (probability), and its level of damage (consequences).

Risk Achievement Worth (RAW): Risk Achievement Worth (RAW) of a modeled plant feature (usually a component, train, or system) is the increase in risk if the feature is assumed to be failed at all times. It is expressed in terms of the ratio of the risk with the event failed to the baseline risk level.

Risk Reduction Worth (RRW): Risk Reduction Worth (RRW) of a modeled plant feature is the decrease in risk if the feature is assumed to be perfectly reliable. It is expressed in terms of the ratio of the baseline risk level to the risk with the feature guaranteed to succeed. See Fussell-Vesely Importance.

Source Term: The radiological source term for a given accident sequence or release category consists of the release fractions for various radionuclide groups (expressed as fractions of initial core inventory), and the timing, elevation, and energy of the release.

Split Fraction: A split fraction is a unitless parameter (i.e., probability) used in quantifying an event tree. It represents the fraction of the time that each possible outcome, or branch, of a particular top event may be expected to occur. Split fractions are, in general, conditional on precursor events. At any branch point, the sum of all the split fractions representing possible outcomes should be unity. (Popular usage equates "split fraction" with the <u>failure</u> probability at any branch [a node] in the event tree.)

Target:
Nuclear Power

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