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amendment

Development of Dry Cask Risk Tools

A Risk Assessment Tool for Reviewal of License Action Requests

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FIGURES

TABLES

ACRONYMS

BWR	Boiling water reactor
CFR	Code of federal regulations
CoC	Certificate of Compliance
FSAR	Final safety analysis report
IFBA	Integral fuel burnable absorber
INL	Idaho National Laboratory
ISFSI	Independent spent fuel storage installation
LAR	License action request
MPC	Multi-purpose Canister
NRC	Nuclear Regulatory Commission
NWTRB	Nuclear Waste Technical Review Board
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
RAI	Request for Additional Information
SDP	Significance determination process
SER	Safety evaluation report
SSC	Structures, systems, and components
SNF	Spent nuclear fuel
SME	Subject matter expert
TMI	Three Mile Island
TS	Technical specifications

1. INTRODUCTION

The NRC has contracted with Idaho National Laboratory (INL) for a project to develop a tool (or methodology) that incorporates information from dry cask Probabilistic Risk Assessments (PRAs) and related available information, to provide the NRC insights in identifying levels of risk at various stages in the nuclear waste path. The project consists of three tasks:

Task 1: This task requires the development of a tool that supports the evaluation of risk related to licensing action requests (LAR) for SNF dry cask storage systems.

Task 2: This task includes the expansion of the tool to spent nuclear fuel (SNF) transportation systems.

Task 3: This task addresses eventual additional regulatory applications (e.g., oversight) for dry cask storage or transportation.

This report is a working document. At the current stage, it provides an overview on INL's work on the first task, but eventually, the document will be further developed in the future to cover INL's work on the remaining two tasks.

The definition of the first task of this project specifically asked for the development of a methodology that incorporates available risk insights and supports the process of identification and prioritization of LAR reviews related to SNF dry storage applications. Fundamentally, the methodology (i.e., tool) should support the NRC staff's review by providing a basic, risk-informed framework that can be used to define recommendations of the depth and breadth of LAR reviews. The staff's review of a licensing action can involve a variety of issues and safety concerns that can vary from routine (e.g., an amendment that is very similar to a recent approval) to complicated (e.g., a licensing action that includes a novel design or materials, or a significant reduction in safety margins). The staff use judgement and experience in their initial review when estimating the level of effort necessary (e.g., the rigor of the review) and the potential need for additional information from the applicant. Staff consideration includes the complexity and quality of the application, the introduction of new design standards, methods of evaluation and boundary conditions, new designs, margins and uncertainties, experience from prior operations and licensing actions, and the potential for the cumulative effects of several changes to create a higher risk than that identified for each individual change. The risk tool represents a resource for the staff to use to further inform staff reviews by providing risk information based on a review of available risk assessments and a sample of past licensing actions and staff experiences. This will allow the staff to better focus on the potentially more risk significant aspects of a licensing action.

The tool uses quantitative, qualitative, and/or semi-quantitative approaches to define a preliminary risk estimate of a requested change (i.e., an LAR item). The present tool design, which was selected by INL in consultation with NRC, is a tree diagram methodology including an associated technical report (i.e., the present document) that summarizes the evaluation background, and outlines user guidance for the tool applications. The structure of the present tool allows NRC reviewers to efficiently conduct a preliminary risk determination of typical LAR items, such as dry storage system design changes, changes in the approved SNF content or

evaluation methods, or editorial changes. Further, the tool provides the user with a rationale behind the risk estimation of each specific change. The risk estimation could be used to define specific, actionable review recommendations for each individual LAR item. This could allow for a more consistent and review process by the NRC as well as for an improvement of the overall efficiency of the review itself.

It is important to emphasize that the risk significance estimations provided by the tool are adaptable and can be changed by the user to address LAR item-specific or dry storage system-specific details or updated guidance. The user of this tool must be aware that the risk posed by an LAR item could be significantly higher (or lower) than outlined in this document due to specific details or combinations of multiple individual LAR items which cannot be caught in a general assessment. As a result, the benefit in using the risk tool is that it provides a database of current risk information that staff should use to 1) consider in deciding on a review strategy and development of Requests for Additional Information (RAIs), 2) allow for discussion of risk significance with other staff reviewers, and 3) use of risk significance to be more effectively applied in licensing decisions and documentation. The risk tool is to be considered a ‘living’ document that will be updated as staff’s understanding and experience increases. Thus, the risk significance in the risk tool is expected to evolve over time such that it will continue to exist as the ‘current’ understanding to be used by staff today and in the future.

2. METHODOLOGY

2.1 Tool Structure

The general approach to tackling the first task of this project was to develop a tree diagram (which can be found in the Appendix of this document, see Figure A.1) that provides the user with preliminary estimations of the risk significance of LAR items. Subsequently, this information can be used (e.g., by a reviewer) to quickly estimate the level of effort and resources required to properly assess an applicant's LAR and provide the basis for moving forward with a risk-informed review. Importantly, this is an initial step, and the review approach and risk significance can change as the review progresses. The tree diagram is accompanied by a rationale document (i.e., the present report) to establish the criteria that were used to estimate the risk significances of a number of potential changes. Users are encouraged to study the applicable rationale so they can assess whether the estimated risk is adequate for a specific LAR item or if additional considerations should be taken to address any relevant LAR item-specific or dry storage system-specific characteristics, as well as effects related to the combination of multiple individual changes.

License action requests for dry storage systems, submitted by the Certificate of Compliance (CoC) holder or licensee to U.S. NRC, can include a request for a single change, or requests for multiple individual changes (e.g., a design change in addition to editorial changes in the technical specifications [TSs]). Each individual change must be reviewed on its own to assess the specific level of risk associated with the change, although the potential cumulative effect of multiple changes must also be considered, as noted below.

Based on past LARs, and in coordination with NRC project management, a set of common, individual LAR items was identified and is evaluated within this tool, while providing the associated risk estimation, including rationale. However, the list of LAR items cannot cover all possible future changes, since situations may arise where the user of the tool wants to evaluate an item that was not yet evaluated in the report. In such situations, the tool provides guidance for a general risk assessment approach that allows the user to estimate the risk associated with an unevaluated LAR item according to the identical criteria used by the tool developers. The tool structure was designed for growth, and the tree diagram and the associated rationales can be extended by additionally identified LAR items in the future. Further, the design allows for risk estimates to be changed to consider new or additional data or risk insights.

Note that this tool is designed to evaluate individual changes. License action request reviewers should be aware of the aggregated level of risk due to the cumulative effect of multiple individual LAR items. In such situations, the reviewer will need to decide whether or not an adapted depth of review is necessary to accommodate the total level of risk. An LAR needs to be considered within the context of the integrated system that it is part of, both in terms of the technical system and the regulatory system that is in place to ensure safety.

The rationales of the LAR items offer NRC reviewers brief discussions on the critical component functionalities and point out potential concerns or safety issues. However, the risk estimation criteria used to establish the rationales consider only the main safety functions of a dry storage system—confinement, radiological protection, and criticality safety in a relative sense (i.e., the significance of a specific component for affecting the safety function). The criteria do not focus on meeting regulatory thresholds or requirements (e.g., a specific dose limit).

2.2 Definitions

A variety of different dry storage systems with significantly different designs are currently used in the U.S., and new systems might be deployed in the future. Some of them use metal casks as the outermost shell, while others use different material, such as concrete overpacks. Some are designed to store SNF in a welded inner canister, while other systems are designed as bare fuel casks. This list of individual characteristics is not exhaustive as it does not cover all possible changes that a reviewer may encounter. Additionally, package operations, such as loading and transfer, are often nuclear power plant specific. Also, different transfer cask designs are used. However, despite the variety of existing designs of the components used in a dry storage operation, the individual component functionalities are often similar or even identical.

To improve the applicability of the presented tool on a large variety of different designs, the categorization of the LAR items did not, whenever possible, consider design-specific individual components. But users of the tool are encouraged to consider these details in their assessments. The component categorization used in the tool structure loosely follows the categorization as outlined in NUREG/CR-6407 (U.S. NRC 1996), and the authors refer to this document's component definitions. Nevertheless, the following list provides a summary of the main dry storage components, including a brief description of their safety-related functions as considered in this tool.

Outer Shell/Overpack: If the outer cask shell is not part of the confinement barrier itself, the outer shell (typically made of metal) or overpack (typically made of concrete) of a dry storage system functions as physical protection of the confinement barrier from mechanical and thermal loads, and provides radiological protection of the environment (i.e., shielding).

Canister/Inner Shell: The welded canister or inner shell of a bolted cask represents e confinement barrier of a dry storage system. The SNF is placed inside the canister or inner shell. The functional difference between a canister-based system and a bolted cask system is that the canister is moveable, but the inner shell of the bolted cask is permanently connected to the cask.

Transfer Cask: The transfer cask is used in the loading and transfer phase of dry storage operations. It is used to move the welded canister or the bare SNF from the spent fuel pool to the overpack or underground enclosure. It provides physical protection of its content, as well as radiological protection through shielding (e.g., through a water jacket).

Basket: The basket is a cell-shaped structure placed in the canister or inner cask. It provides criticality control by preventing SNF assemblies, which are placed in the basket cell, from deforming or relocating excessively.

2.3 Tree diagram

The full tree diagram is displayed in Figure A.1 in the Appendix, but an example portion of the tree diagram is shown in Figure 1. The diagram is designed for ease of use and allows the reviewer to quickly identify the risk estimate for a specific LAR item.

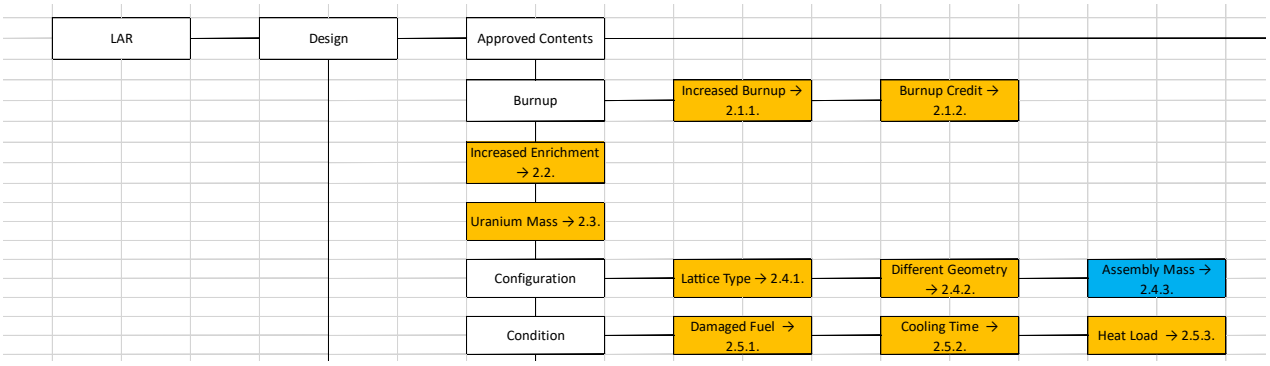

Figure 1. Example portion of Task 1 tree diagram.

Each field (i.e., "Gate") in the tree diagram represents an LAR item, and is color coordinated to signal the associated risk significance. Each gate is numbered to help the user to identify the associated rationale, allowing review of the discussion bases used in the risk classification.

The risk categorization key shown in Table 1 is color coded for easy recognition, and outlines the specific criteria according to which change in the tree diagram is evaluated and categorized. The column "LAR Review Recommendations" in Table 1 is designed to be modified or expanded as the NRC uses the tool and defines the level of review that would be adequate for the respective risk category.

Table 1. Risk categorization key.

Criterion: / Risk Significance	A: Factor of Risk Increase*	B: Effect on current/future operations	C: Redundancy	D: Safety margins	E: Evaluation Complexity	F: Flaw/Error Detection Probability	LAR Review Recommendations
Low	< 2 or a Decrease in Risk	Yes or No	Exists	Small to Large	Simple	High	Efficient
Medium	≥ 2 and < 10	Yes	Nonexistent	Large	Simple to Complex	Medium to High	In Detail
High	≥ 10	Yes	Nonexistent	Small	Simple to Complex	Low to High	Extensive, Thorough, Very Detailed
See Rationale	No risk significance estimation or review recommendations possible without consideration of additional factors.						

*This factor describes the increase in canister release risk due to an LAR item, compared to the baseline release risk calculated for a dry storage operation described in NUREG-1864. This criterion is only applicable If a quantitative risk sensitivity study is possible.

Table 2 provides a set of six rules the user must follow to ensure no change is screened out or remains unanalyzed.

Table 2. Dry storage cask risk tool user guidance.

Rule	Instructions
1	Choose the gate that matches the LAR item.
2	If more than one choice of a gate is possible, evaluate both gates independently.
3	Use color scale to evaluate risk significance.
4	Review rationale document if gate color is blue.
5	Review rationale document for general background information or if there are doubts.
6	Review rationale document if the gate is unevaluated.

2.4 Risk Significance Estimations

The following sections outline the bases for the LAR risk significance determination tool visualized in the accompanying tree diagram. The determination of the risk significance of the evaluated LAR items is based on semi-quantitative (i.e., considering available simulation or examination data, and sensitivity studies on PRA data), and qualitative assessments (i.e., via conducting SME interviews, and engineering judgment) of the potential consequences of an LAR item (e.g., a design change of a component) on system safety, focusing on the following core functions of a SNF dry storage systems:

1. Confinement of radionuclides
2. Radiological protection of the public and operating personnel (i.e., shielding capability of the system)
3. Criticality safety (i.e., subcriticality of contents)

It is recognized that some of the studies and evaluations cited use specific designs and assumptions in estimating risks. The information in this report does not represent an endorsement of regulatory acceptance of any particular dry storage system design or assumption. The risk significance estimates provide information to assist staff reviews and represents an initial starting point that will be used to help inform the start of a review—importantly it is understood the specific review represents the licensing position and could end with an understanding of significance that is different from what was initially estimated in this risk tool. As appropriate, it is expected that revisions to the risk tool will be made to improve its reasonableness and effectiveness.

2.4.1 Evaluated Changes

The evaluated dry storage system LAR items listed in this document are a set of generic items selected from common, past LARs submitted to U.S. NRC, or were selected in coordination with U.S. NRC project management. The focus of the evaluations was dry storage design changes, changes regarding the chosen safety evaluation methods, editorial changes, and changes in the approved SNF content of the dry storage system. However, it is impossible to develop (and evaluate) a complete set of LAR items since future requests cannot be adequately

predicted. Thus, a strategy is provided (see Section 2.5.7) to evaluate LAR items that are not included within this document.

It is important to note that this risk tool did not take into account the effects of long-term material degradation, such as stress corrosion cracking. However, such effects could be included by expanding the tree diagram by a branch that is focused on LAR items related to license extensions. The expandable structure of the tool allows for a permanent incorporation of additional LAR items.

2.4.2 Risk Significance Evaluation Criteria

The risk significance of an LAR item was estimated by evaluating if, and under which circumstances, the requested change could significantly increase the likelihood and consequence of a release of radioactive material to the environment, dry storage content criticality, or exposure of the public or operating personnel to radiation. The risk categorization key (i.e., the evaluation criteria) used in the evaluation process is presented in Table 1. The risk significance estimation included quantitative information (if available), deliberations on the effect of the change on the present or future dry storage procedures, the level of redundancy of the system of preventing the initiation of an accident (in using this tool, reviewers should consider not only redundancy of the specific component but also if there are different components that provide redundancy of the function), safety margin estimates, the evaluation complexity of the LAR item, and an evaluation of the likelihood of the detection of a flaw or an error in the review process of an LAR item or during operations. The last column in Table 1 provides a basic review recommendation, associated with the risk significance estimation of an LAR item.

Further, the risk evaluations are meant as a first, preliminary estimation of the risk associated with an LAR item and are intended to support the NRC reviewers in planning and conducting the review. It is not meant to be a static risk estimate. During review and under consideration of LAR item-specific and dry storage system-specific details, the reviewer could gain the impression that the risk significance could be different from the estimate provided by this tool, and the review process of the LAR item should be adapted accordingly. This process is visualized in the flow chart of Figure 2

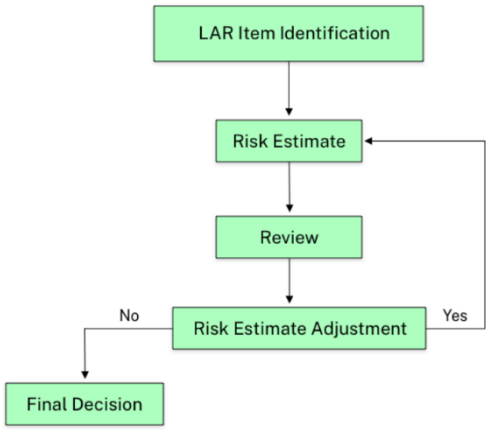

Figure 2. Review flowchart.

2.4.3 Available Quantitative Data and Literature

Table 3 shows a summary of a PRA data for a welded dry cask storage system published in NUREG-1864 (U.S. NRC 2006). The PRA evaluates the risk of a dry storage operation with specific characteristics, but its results provide valuable insights, and its data can be studied to obtain a general sense of risk related to a variety of relevant design changes. The presented studies on the quantitative risk data are based on the NUREG-1864 PRA and it is important to note that they only meant to support the risk significance estimation process of a specific LAR item, rather than determining it. For the determination, additional evaluation criteria, such as system redundancies, or safety margins, were taken into account (see Table 1).

In the scope of the NUREG-1864 PRA, risk is defined as the probability of latent cancer fatality for an individual within a 10-mile radius around the dry storage facility due to the release of noble gases or radioactive isotopes, under consideration of a 20-year dry storage operation duration. The PRA was conducted to investigate the probability of a release of noble gases or radioactive isotopes for the different phases of a dry storage operation. The studied initiating events included SNF assembly or canister drops, besides external hazards such as earthquakes or airplane impacts. The associated risk is 2.4E-12, which is an accumulated value of the demand-based frequency of the initiating events during the handling and transfer phase that occur only once (in the first year of the operation), and the time-based annual frequency of the initiating events during the 20-year dry storage phase.

The probabilistic data of NUREG- 1864 is used to study the risk sensitivity to applicable LAR items. However, instead of using the cancer risk as a unit of measure, the increase in risk of release of radioisotopes, such as SNF fragments (fuel fines), noble gases, or crud, from the canister release risk (for a 20-year dry storage operation) was evaluated. This value was computed as the summation of the products of the accident probability in a specific dry storage operation phase (marked red in Table 3) and the probability of a release of radioactive isotopes (or fuel fines), crud, or noble gases (marked blue in Table 3). An example computation for the NUREG-1864 PRA dry storage system operation is presented in Table 4. The baseline canister release risk of the unmodified dry storage system is 2.29E-04. This value can be compared to the risk of a modified system, as demonstrated in some of the rationales in this document where, for the purposes of estimating the risk significance, the assumption was made that a specific component does not properly perform its design safety function.

Table 3. Cancer risk of a 20-year dry storage operation, adapted from NUREG-1864 PRA data.

Phase	Stage	Activity	Event	Initiating Event Frequency	Probability of Release from Canister and Rod	Released Material Type	Probability of Release to Environment	Consequences	Risk
Handling	1	Load SNF in Canister	Fuel Assembly Drop	2.20E-03	6.40E-02	Noble Gas	1.00E+00	1.50E-12	2.11E-16
	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	Noble Gas	1.00E+00	1.00E-10	5.60E-15
	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	5	Moving Transfer Cask to Preparation Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	6	Moving Transfer Cask to Preparation Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	7	Moving Transfer Cask to Preparation Area, 3rd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	12	Moving Transfer Cask to Bottom Lid Exchange Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	13	Moving Transfer Cask to Bottom Lid Exchange Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	15	Moving Transfer Cask in Direction Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	All	1.50E-04	3.60E-04	3.02E-18
	18	Lowering Transfer Cask onto Overpack	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	Noble Gas	1.00E+00	1.00E-10	1.12E-16
						Other than Noble Gas	1.50E-04	3.60E-04	6.05E-14
	20	Lifting Canister and Opening Transfer Lid	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	Noble Gas	1.00E+00	1.00E-10	1.57E-15
						Other than Noble Gas	1.50E-04	3.60E-04	8.47E-13
	21	Transferring Canister to Overpack	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	Noble Gas	1.00E+00	1.00E-10	1.57E-15
						Other than Noble Gas	1.50E-04	3.60E-04	8.47E-13
								Risk Handling Phase (First Year Only)	**1.76E-12**
Transfer	26	Lifting Storage Overpack from Helman Rollers with Overpack Transporter	Dropping Overpack	0.00E+00	0.00E+00	All	0.00E+00	1.00E-10	0.00E+00
	27	Moving Storage Overpack to Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	All	0.00E+00	1.00E-10	0.00E+00
	28	Holding Storage Overpack	Dropping Overpack	0.00E+00	0.00E+00	All	0.00E+00	1.00E-10	0.00E+00
	29	Moving Storage Overpack away from Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	All	0.00E+00	1.00E-10	0.00E+00
	30	Transferring Storage Overpack on Asphalt to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	All	0.00E+00	1.00E-10	0.00E+00
	31	Transferring Storage Overpack on Gravel to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	All	0.00E+00	1.00E-10	0.00E+00
	32	Moving Storage Overpack above Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	All	0.00E+00	1.00E-10	0.00E+00
	33	Lowering Storage Overpack onto Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	All	0.00E+00	1.00E-10	0.00E+00
								Risk Transfer Phase (First Year Only)	**0.00E+00**
Storage	34	Dry Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	All	1.00E+00	3.60E-04	2.52E-16
			Overpack Struck by Aircraft	6.30E-09	1.40E-02	All	1.00E+00	3.60E-04	3.18E-14
			Overpack Struck by Meteorite	3.50E-14	1.00E+00	All	1.00E+00	3.60E-04	1.26E-17
			Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	All	1.00E+00	3.60E-04	0.00E+00
								Risk Storage Phase (Per Year)	**3.20E-14**
								Total Risk (20 Years)	**2.40E-12**

Table 4. Baseline canister release risk of a 20-year dry storage operation data, adapted from NUREG-1864 PRA data.

Phase	Stage	Activity	Event	Initiating Event Frequency	Probability of Release from Canister and Rod	Release Risk
Handling	1	Load SNF in Canister	Fuel Assembly Drop	2.20E-03	6.40E-02	1.41E-04
	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	5	Moving Transfer Cask to Preparation Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	6	Moving Transfer Cask to Preparation Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	7	Moving Transfer Cask to Preparation Area, 3rd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	12	Moving Transfer Cask to Bottom Lid Exchange Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	13	Moving Transfer Cask to Bottom Lid Exchange Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	15	Moving Transfer Cask in Direction Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	18	Lowering Transfer Cask onto Overpack	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
	20	Lifting Canister and Opening Transfer Lid	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
	21	Transferring Canister to Overpack	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase (First Year Only)	**2.29E-04**
Transfer	26	Lifting Storage Overpack from Helman Rollers with Overpack Transporter	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	27	Moving Storage Overpack to Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	28	Holding Storage Overpack	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	29	Moving Storage Overpack away from Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	30	Transferring Storage Overpack on Asphalt to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	31	Transferring Storage Overpack on Gravel to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	32	Moving Storage Overpack above Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	33	Lowering Storage Overpack onto Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase (First Year Only)	**0.00E+00**
Storage	34	Dry Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
			Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
			Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
			Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (Per Year)	**8.89E-11**
					Total Risk (20 Years)	**2.29E-04**

2.4.4 General Assumptions

For evaluations that consider the sensitivity studies on the probabilistic data provided by NUREG-1864 (U.S. NRC 2006), the following assumptions are made:

1. The integrity of the canister of the dry storage system under consideration is comparable to the integrity of a welded canister that is part of a dry cask storage system analyzed in NUREG-1864.
2. The integrity of the transfer cask of the dry storage system under consideration is comparable to the integrity of the transfer cask of the system analyzed in NUREG-1864.
3. The integrity of the storage overpack of the storage system under consideration is comparable to the integrity of the overpack analyzed in NUREG-1864.
4. The potential increase in the likelihood of canister release is considered to result in an equal increase in the risk to the public or operating personnel.

Other references used in the scope of the risk evaluations presented within this document include a human reliability analysis for spent fuel handling (U.S. NRC 2012a), and the classification of Structures, systems, and components (SSCs) according to their importance to safety as outlined in NUREG/CR-6407 (see Table 5) (U.S. NRC 1996), among others.

Table 5. Categorization of SSCs according to safety importance. Adapted from NUREG/CR-6407 (U.S. NRC 1996).

Category	Criteria
A – Critical to safety	Failure of SSCs could directly lead to loss of confinement, shielding, or criticality control.
B – Major importance to safety	Failure of SSCs in conjunction with failure of another item could lead to loss of confinement, shielding, or criticality control.
C – Minor importance to safety	Failure of SSCs likely does not affect the public health and safety adversely.

2.4.5 Gates – Background

This document accompanies a tree diagram that displays potential LAR items for dry storage systems as numbered gates. The gate color provides information on the risk significance that was estimated for a change (i.e., an LAR item) in the system. The gate number in the tree refers to a section in this document that provides a rationale to the risk significance determination of each change.

2.4.6 Risk Significance

Each gate in the tree diagram is marked with a color that indicates the risk significance of the corresponding change related to core dry storage system functions (i.e., confinement of radionuclides, shielding, and subcriticality of contents). Three levels of risk significance are used in the tool (Table 6):

High: Red-colored gates

Medium: Yellow-colored gates

Low: Green-colored gates

The risk significance of some LAR items cannot be estimated without consideration of LAR item-specific or system design-specific details. The corresponding gates are marked in blue color (Table 6).

Table 6. Risk significance levels of dry storage risk tool.

Risk Significance
Low
Medium
High
See Rationale

The gray gate (9.) in the tree diagram is a gate that can be used to evaluate LAR items not specifically addressed in this tool. This gate refers the user to the risk categorization key presented in Table 1, which provides the risk estimation criteria A-F. The methodology allows the user to make their own initial assessment of the risk related to a currently unevaluated LAR item by analyzing the specific item according to the factor of risk increase (if quantitative data is available), the effect on current and future operations, system redundancies, available safety margins, expected LAR item evaluation complexity, and the risk of missing a design flaw or evaluation error.

2.5 Rationale

The following sections provide the rationales of the risk significance determination of changes in dry storage systems. Each rationale addresses one of the numbered gates in the accompanying tree diagram.

2.5.1 Design – Canister/Inner Cask

Gate 1.1.1.1.1. Shell Body

The shell body denotes the main part of the canister or the inner cask. Its task is to provide confinement of SNF in a dry storage system. This part of a dry storage system is rated as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the canister, the inner cask, or any of their individual parts could directly lead to a release of radioactive material to the secondary containment building (i.e., the containment building in which a dry storage operation takes place) or the environment. Therefore, the shell body is an SSC critical to safety.

This conclusion can be evaluated using available NUREG-1864 PRA (U.S. NRC 2006) data. The data indicates SNF handling as the critical operational phase, with respect to release risk of radioactive isotopes. This phase contributes significantly to the overall risk of a dry storage operation because the canister remains open during the initial SNF handling stages (Stages 1 to 8). The risk of stage 1 is controlled by the risk of SNF assembly failure when dropped during canister loading. A drop of the transfer cask into the cask pit (Stage 3) contributes further to the release risk because this initiating event could cause a release of noble gases. Although the canister is unlikely to fail in stages 4 to 8, due to the low drop height of less than 1 foot, NUREG-1864 considers a canister release probability of 1.00E-6 related to canister weld failure (see Table 4).

Modifying the canister design could increase the risk of canister or canister weld failure, before or after closure, under mechanical or thermal loading. Increasing the canister failure probability in accident scenarios other than SNF assembly handling (i.e., dropping the canister-containing transfer cask) could increase the canister release risk.

A reevaluation of the available PRA data assuming unrealistic canister or canister weld failure indicates an increase in risk for the considered dry storage operation by a factor of almost 5 (see Table 7). This points to a medium risk significance of an LAR item that affects an important component of the confinement barrier, when evaluated according criterion A in Table 1. However, no arguments for large safety margins of the confinement barrier can be made, and there is no redundant confinement function. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of the tool user. The risk significance is determined high.

Table 7. Canister release risk increase assuming an increased canister failure probability due to an improperly modified shell body (Gate 1.1.1.1.1.).

Phase	Stage	Activity	Event	Initiating Event Frequency	Probability of Release from Canister and Rod	Release Risk
Handling	1	Load SNF in Canister	Fuel Assembly Drop	2.20E-03	6.40E-02	1.41E-04
	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	5	Moving Transfer Cask to Preparation Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	6	Moving Transfer Cask to Preparation Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	7	Moving Transfer Cask to Preparation Area, 3rd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	12	Moving Transfer Cask to Bottom Lid Exchange Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	13	Moving Transfer Cask to Bottom Lid Exchange Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	15	Moving Transfer Cask in Direction Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	18	Lowering Transfer Cask onto Overpack	Dropping Transfer Cask, Height 100 ft	5.60E-05	1.00E+00	5.60E-05
	20	Lifting Canister and Opening Transfer Lid	Dropping Canister, Height 5.9 ft	5.60E-05	1.00E+00	5.60E-05
	21	Transferring Canister to Overpack	Dropping Canister, Height 5.9 ft	5.60E-05	1.00E+00	5.60E-05
					Risk Handling Phase (First Year Only	**2.29E-04**
Transfer	26	Lifting Storage Overpack from Helman Rollers with Overpack Transporter	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	27	Moving Storage Overpack to Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	28	Holding Storage Overpack	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	29	Moving Storage Overpack away from Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	30	Transferring Storage Overpack on Asphalt to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	31	Transferring Storage Overpack on Gravel to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	32	Moving Storage Overpack above Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	33	Lowering Storage Overpack onto Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase (First Year Only)	**0.00E+00**
Storage	34	Dry Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E+00	7.00E-07
			Overpack Struck by Aircraft	6.30E-09	1.00E+00	6.30E-09
			Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
			Overpack Headed by Aircraft Fuel	3.70E-09	1.00E+00	3.70E-09
					Risk Storage Phase (Per Year)	**8.89E-11**
					Total Risk (20 Years)	**2.29E-04**
					Risk Increase Factor	**4.58E+00**

Gate 1.1.1.1.2. Bottom Head

The bottom head of the canister or inner cask is part of the confinement barrier of a dry storage system. Thus, it is rated as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the confinement barrier or any of its individual parts could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.1.3. Top Head

The top head of the canister or inner cask is part of the confinement barrier of a dry storage system. Thus, it is rated as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the confinement barrier or any of its individual parts could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.2.1. Lid Design

The top head of the canister or inner cask is part of the confinement barrier of a dry storage system. Thus, it is rated as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the confinement barrier or any of its individual parts could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the

confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.2.2. Lid Seal

The lid seal of the canister or inner cask is part of the confinement barrier of a dry storage system. Thus, it is rated as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the confinement barrier or any of its individual parts could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.2.3. Closure Hardware

The closure hardware of the canister or inner cask is part of the confinement barrier of a dry storage system. Thus, it is rated as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the confinement barrier or any of its individual parts could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.3.1. Baseplate Design

The base plate of the canister or inner cask is part of the confinement barrier of a dry storage system. Thus, it is rated as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the confinement barrier or any of its individual parts could lead directly to release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential

for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.4.1. Welding Material

The welding material of the canister or inner cask can affect the integrity of the confinement barrier of a dry storage system and is therefore rated a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the confinement barrier or any of its individual welds could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.4.2. Welding Method

The welding methods use in them manufacturing process of the canister or inner cask can affect the integrity of the confinement barrier of a dry storage system. A failure of the confinement barrier or any of its individual welds could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.5.1. Vent/Drain Port

The vent and drain ports of the canister or inner cask, including accompanying seals and plugs, are part of the confinement barrier of a dry storage system. Thus, they are rated as Category A components according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the confinement barrier or any of its individual parts could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building of the nuclear

power plant) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.5.2. Leak Check Port

The leak check port of the canister or inner cask, including accompanying seals, is a primary part of the confinement barrier of a dry storage system. The leak check port plug provides a second layer of confinement and may be necessary to provide sufficient shielding for appropriate radiological protection to the operating personnel (U.S. NRC 1996).

The confinement barrier and the leak check port plug of a canister are rated as Category A components according to NUREG/CR-6407 (U.S. NRC 1996). A failure of the barrier or any of its individual parts could lead directly to a release of radioactive material into the secondary containment building (i.e., the containment building in which a dry storage operation takes place) or to the environment. Therefore, these parts are SSCs important to safety. Using available NUREG-1864 PRA data (U.S. NRC 2006), this conclusion can be evaluated by studying the potential effect on the release likelihood if this component did not perform its safety function as a confinement barrier. These evaluations indicate the potential for an increase in the likelihood of a release by a factor of almost 5 assuming unavailability of the safety function (see rationale of Gate 1.1.1.1.1). Thus, a medium risk significance would be indicated for such a modification. However, if no arguments for large safety margins of the confinement barrier can be made, the risk significance is determined to be high. Any LAR item related to a design change of the confinement barrier requires the full attention of the reviewer.

Gate 1.1.1.6. Backfill Pressure

The backfill pressure in a dry storage system is typically designed to reach pressures in the order of 0.5 MPa (72.5 psi) or less after reaching equilibrium at the beginning of storage. A common backfilling gas is helium. A modification of the helium backfill pressure requires a thermal reevaluation of the system to ensure necessary thermal heat transfer of the SNF, and to avoid over-pressurization of the system. An over-pressurization could, although unlikely, lead to a leaking confinement and, consequently, to a release of radioactive material to the environment. Although no confinement barrier redundancy exists in a typical SNF dry storage system design, multiple adverse conditions need to happen at the same time to enable such a scenario. For instance, rod cladding failure is required to increase the internal pressure. In addition, a fire scenario could also result in over pressurization. In such extreme event, there may only be a small margin relative to operating pressure and design allowable pressure.

Based on the discussion summarized above, the risk significance of a modification of the backfill pressure is qualitatively evaluated as medium.

Gate 1.1.2.1. Basket

The basket is the mechanical support of the SNF in place within the canister or cask. It prevents the fuel and neutron absorber from excessively deforming and relocating. The spacing of the SNF assemblies and neutron absorbers is part of the criticality control of the dry storage system. The basket is classified as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996).

Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215 dry storage canisters showed that the content of most canister remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as SNF burnup) in the original, upper-bound criticality evaluations as published in the final safety analysis reports (FSARs) of dry storage systems. The United States Nuclear Waste Technical Review Board (U.S. NWTRB) estimates the likelihood of criticality in an SNF package as low, even if the cask is fully flooded or the neutron absorbers are removed or degraded (Rigby 2010). Further, Alsaed (2018) suggests that without significant moderation, commercial SNF cannot reach criticality. Nevertheless, the analyses of K. Banerjee (2015) indicate that the probability of criticality of a fully flooded canister increases if the basket is fully degraded, allowing the SNF assemblies to freely relocate.

The available literature and assessments lead to the qualitative determination that the risk significance due to a modification of the basket structure is low. Criticality in a dry storage canister can only occur after exceeding significant criticality safety margins, and concurrence of multiple adverse conditions, requiring the failure of the independent confinement barrier.

Gate 1.1.2.2. Neutron Absorber

Some dry storage canisters contain neutron absorbers for criticality control of the content. The absorbers are classified as Category A components according to NUREG/CR-6407 (U.S. NRC 1996).

Although no quantitative PRA data on cask subcriticality is available, recent evaluations of 215 dry storage canisters showed that their content remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. The U.S. NWTRB estimates the likelihood of criticality in an SNF package is low, even if the cask is fully flooded or the neutron absorbers are removed or degraded (Rigby 2010). This data supports the assumptions that criticality safety margins exist, and that the occurrence of multiple adverse conditions, such as a failure of the independent confinement barrier, is necessary to reach criticality in a dry storage canister.

The available literature and data lead to the qualitative determination that the risk significance of a modification of neutron absorbers with respect to the original design is low.

Gate 1.1.3.1. Shield Plug

Some canister designs include a shield plug placed at the top end of the canister to protect operating personnel from radiation exposure during welding of the canister lid, or when moving the canister into the storage overpack, before closing the shielded storage overpack lid or door.

The shield plug is classified as a Category A component according to NUREG/CR-6407 (U.S. NRC 1996). There is no redundancy incorporated in a typical dry storage design, and thus, an improper modification of the shield plug could lead to a significantly increased dose rate received by the operating personnel should the shied plug no longer perform its safety function. Further, the safety margins are likely lower than the safety margins for the general radiological protection provided by the shielding of the cask, although it is important to recognize that normally, the radiation levels during loading are actively measured.

However, no justification can be made for an estimation of a lower risk significance for an LAR item that includes a shield plug modification, and the qualitative risk significance determination for such a modification is estimated as high.

Gate 1.1.4.1. Canister Hardware

The canister hardware includes keepers, bolts, nuts, cotter pins, detent pins, lockwires, or lanyards that prevent bolts and nuts from loosening or from being incorrectly assembled. Hardware parts are classified as Category C components according to NUREG/CR-6407 (U.S. NRC 1996).

The function of the canister hardware is limited to structural integrity (U.S. NRC 1996). A modification of the canister hardware does not have an effect on the core safety functions directly of the system (i.e., shielding, confinement, or criticality control). Thus, the qualitative determination for the risk significance of canister hardware modification is low.

Gate 1.1.5.1.1. Canister Lugs/Trunnions/Grapples

Lugs, trunnions, and grapples of the inner canister of a dry storage system are important for successful canister lifting and moving operations. These components are classified as Category A items according to NUREG/CR-6407 (U.S. NRC 1996). Although listed under operations support, the reliability of lugs, trunnions, and grapples, including their bolts or welds, affect the risk of release of radioactive material from the canister to the environment. Specifically, these components play an important role during canister transfer from the transfer cask into the storage overpack, and their design could have a controlling effect on the initiating accident frequency in the corresponding SNF handling stages (i.e., Stage 20 and 21 in Table 4). The baseline NUREG-1864 PRA data (U.S. NRC 2006) indicate a canister lifting accident probability of 5.6E-5. A component failure due to a flawed design could significantly increases the likelihood of a canister drop during handling operations (see Table 8).

Based on this evaluation, the risk significance of an LAR item that includes a modification of the canister lugs, trunnions, or grapples, is determined as high.

Table 8. Canister release risk increase assuming an increased canister failure probability due to improperly modified canister lugs, trunnions, or grapples (Gate 1.1.5.1.1.).

Phase	Stage	Activity	Event	Initiating Event Frequency	Probability of Release from Canister and Rod	Release Risk
Handling	1	Load SNF in Canister	Fuel Assembly Drop	2.20E-03	6.40E-02	1.41E-04
	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	5	Moving Transfer Cask to Preparation Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	6	Moving Transfer Cask to Preparation Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	7	Moving Transfer Cask to Preparation Area, 3rd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	12	Moving Transfer Cask to Bottom Lid Exchange Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	13	Moving Transfer Cask to Bottom Lid Exchange Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	15	Moving Transfer Cask in Direction Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	18	Lowering Transfer Cask onto Overpack	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
	20	Lifting Canister and Opening Transfer Lid	Dropping Canister, Height 5.9 ft	1.00E+00	2.80E-01	2.80E-01
	21	Transferring Canister to Overpack	Dropping Canister, Height 5.9 ft	1.00E+00	2.80E-01	2.80E-01
					Risk Handling Phase (First Year Only	**5.60E-1**
Transfer	26	Lifting Storage Overpack from Helman Rollers with Overpack Transporter	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	27	Moving Storage Overpack to Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	28	Holding Storage Overpack	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	29	Moving Storage Overpack away from Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	30	Transferring Storage Overpack on Asphalt to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	31	Transferring Storage Overpack on Gravel to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	32	Moving Storage Overpack above Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	33	Lowering Storage Overpack onto Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase (First Year Only)	**0.00E+00**
Storage	34	Dry Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
			Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
			Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
			Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (Per Year)	**8.89E-11**
					Total Risk (20 Years)	**5.60E-01**
					Risk Increase Factor	**2.44E+03**

Gate 1.1.5.1.2. Canister Lug/Trunnion/Grapples Welds or Bolts

The welds and bolts of the lugs, trunnions, and grapples of the inner canister of a dry storage system are important for successful canister lifting and moving operations. Modifying the design of the welds or bolts of the lugs, trunnions, or grapples, could affect the reliability of the lugs, trunnions, or grapples itself. These components are classified as Category A items according to NUREG/CR-6407 (U.S. NRC 1996). Although listed under operations support, the reliability of lugs, trunnions, and grapples, including their bolts or welds, affect the risk of release of radioactive material from the canister to the environment. Specifically, these components play an important role during canister transfer from the transfer cask into the storage overpack, and their design could have a controlling effect on the initiating accident frequency in the corresponding SNF handling stages (i.e., Stage 20 and 21 in Table 4). The baseline NUREG-1864 PRA data (U.S. NRC 2006) indicate a canister lifting accident probability of 5.6E-5. A component failure due to a flawed design could significantly increases the likelihood of a canister drop during handling operations (see Table 8).

Based on this evaluation, the risk significance of an LAR item that includes a modification of the canister lugs, trunnions, or grapples, is determined as high

Gate 1.1.6. Canister Type

A new canister model requires a thorough reevaluation of the dry storage system with respect to criticality control, confinement, shielding capability, structural integrity, thermal performance, and operations. Failure of a new canister to perform its safety function would pose a significant, direct risk to the safety of the operating personnel and the public because the canister represents the confinement barrier. Thus, the risk significance of an LAR item that includes the introduction of a new canister type is qualitatively determined as high.

2.5.2 Design – Outer Shell/Overpack

Gate 1.2.1.1. Neutron Shield

A dry storage system needs to provide adequate neutron and gamma shielding to protect the public and personnel from radiation originating from the SNF. Depending on the dry storage system type, neutron and gamma radiation shielding can be achieved by a single concrete overpack structure, or by the combined shielding effect of different layers of shielding materials. In metal cask systems, a highly dense inner shielding material (e.g., lead or stainless steel) provides gamma shielding, followed by a neutron-shielding material (e.g., vinyl ester resin). Additional secondary gamma radiation shielding can be added by the stainless-steel shell as the outermost layer. With some exception for specific dry storage system designs, the neutron shielding of a dry storage system is classified as Category B components (i.e., a failure would be a major safety concern) according to NUREG/CR-6407 (U.S. NRC 1996).

Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The

exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded. Thus, it is expected that radiation exposure to personnel and the public due to a lack of shielding capability would be detected within a time frame that allows corrective measures to be put into place. Nonetheless, the shielding design must be reviewed to ensure proper shielding performance.

Based on the discussion summarized above, the risk significance of a change of shielding components is determined as medium.

Gate 1.2.1.2. Gamma Shield

A dry storage system needs to provide adequate neutron and gamma shielding to protect the public and personnel from radiation originating from the SNF. Depending on the dry storage system type, neutron and gamma radiation shielding can be achieved by a single concrete overpack structure, or by the combined shielding effect of different layers of shielding materials. In metal cask systems, a highly dense inner shielding material (e.g., lead or stainless steel) provides gamma shielding, followed by a neutron-shielding material (e.g., vinyl ester resin). Additional secondary gamma radiation shielding can be added by the stainless-steel shell as the outermost layer. With some exception for specific dry storage system designs, the gamma shielding of a dry storage system is classified as a Category B component (i.e., a failure would be a major safety concern) according to NUREG/CR-6407 (U.S. NRC 1996).

Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded. Thus, it is expected that radiation exposure to personnel and the public due to a lack of shielding capability would be detected within a time frame that allows corrective measures to be put into place. Nonetheless, the shielding design must be reviewed to ensure proper shielding performance.

Based on the discussion summarized above, the risk significance of a change of shielding components is determined as medium.

Gate 1.2.1.3. Shielded Access Door

Some concrete dry storage systems use a shielded access door to close the concrete overpack after loading of the canister. The door shields the environment from radiation originating from the canister. Secondary gamma radiation shielding can be added by the stainless-steel shell as the outermost layer. The shielded access door is classified as Category B component (i.e., a failure would be a major safety concern) according to NUREG/CR-6407 (U.S. NRC 1996).

Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating

from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded. Thus, it is expected that radiation exposure to personnel and the public due to a lack of shielding capability would be detected within a time frame that allows corrective measures to be put into place. Nonetheless, the shielding design must be reviewed to ensure proper shielding performance.

The risk significance of an LAR item that includes a modification of the access door is evaluated as medium.

Gate 1.2.2.1. Fins

Dry storage systems use components such as fins to transfer decay heat produced by the SNF in the cask to the environment. Further, the design of these components affects the cask temperature in a fire accident. NUREG/CR-6407 classifies temperature control components as Category A items (U.S. NRC 1996), which indicates that they play an important role ensuring the safety of a dry storage system.

However, thermal simulations of cask storage systems can provide conservative data (Hanson, et al. 2016). For instance, temperature data recorded during backfilling gas experiments in a CASTOR V/21 cask loaded 1986 at Surry Power Station indicate that the peak cladding temperature of most SNF rods could be significantly below the generally permitted peak cladding temperature of 400 °C. Further, newer simulations and experimental measurements indicate the peak cladding temperatures of high burnup SNF rods likely do not exceed the permitted thresholds, although evaluation uncertainties should be considered (Fort, et al. 2019, Csontos 2020, Dziadosz, et al. 1986, U.S. NRC, 2019, U.S. NRC, 2020).

Nevertheless, the applicant needs to ensure compliance of the peak cladding temperature with the thermal limits, by conducting a thermal assessment of the dry storage system. These temperature thresholds do not represent hard limits to indicate cladding failure or lack thereof. Rather, these temperature thresholds were put into place to prevent cladding from deforming or degrading due to effects like hydride embrittlement or hydride reorientation; recent research (Eidelpes, Ibarra and Medina 2019), however, indicates latter effects might not play as important a role as previously thought.

The effectiveness of fins for temperature control during a fire accident may be very limited, and some cask vendors already consider a loss of neutron shielding during fire in their safety evaluations. Fuel rod failure due to high temperatures alone would not lead to criticality, a release of radioactive material to the environment [for a “leaktight” system, as defined by ANSI N14.5 “American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment” (U.S. ANSI 2014)], or a significant increase in radiation exposure of the public or the operation personnel; however, fuel rod failure affects the dry storage system confinement function. In particular, redundancy due to the confinement barrier provided by fuel cladding would not exist if the safety function of the fins could not be relied on to preclude the potential for high temperatures that could affect both the integrity of the SNF rods as well as the integrity of the canister.

Furthermore, it is important to note that the discrepancy between thermal simulations and empirically collected data suggest a complex evaluation process of a modification of heat removal components, with a significant probability of an error being undetected.

Based on the discussion above, the risk significance of a change in fin design is qualitatively evaluated as medium.

Gate 1.2.2.2.1. Vent Openings

In NUREG-1864 (U.S. NRC 2006), long-term blockage of vents in an overpack was evaluated via simulations of steady-state conditions. This event can be considered a worst-case scenario regarding the effect of a design modification on the heat transfer capability of the dry storage system. The simulations indicate that blocked vents or vent channels would increase the peak fuel rod temperature significantly, and about 50% of all fuel rods could experience peak cladding temperatures above the long-term temperature threshold of 400°C, reaching up to 461°C. The peak canister temperature in the simulation is 283°C. However, the study concluded that the canister would not fail under such conditions. Further, blocked vents or vent channels for a short-term period likely would not lead to fuel rod failure.

As opposed to a short-term event like a blocked vent, a failure in vent design could lead to a long-term thermal issue, and the SNF peak cladding temperature could exceed the long-term temperature limit for a duration longer than the short-term period described above. Fuel rod failure due to high temperatures alone would not lead to criticality, a release of radioactive material to the environment [for a “leaktight” system, as defined by ANSI N14.5 “American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment” (U.S. ANSI 2014)], or a significant increase in radiation exposure of the public or the operation personnel; however, fuel rod failure affects the dry storage system confinement function. In particular, redundancy due to the confinement barrier provided by fuel cladding would not exist if the safety function of the vents could not be relied on to preclude the potential for high temperatures that could affect both the integrity of the SNF rods as well as the integrity of the canister.

Based on the discussion summarized above, although thermal components are classified as Category A components in NUREG/CR-6407 (U.S. NRC 1996), the determination of the risk significance for an LAR item that includes a modification of the vent openings is rated as medium.

Gate 1.2.2.2.2. Vent Channels

In NUREG-1864 (U.S. NRC 2006), long-term blockage of vents in a dry storage overpack was evaluated via simulations of steady-state conditions. This event can be considered a worst-case scenario regarding the effect of a design modification on the heat transfer capability of the dry storage system. The simulations indicate that blocked vents or vent channels would increase the peak fuel rod temperature significantly, and about 50% of all fuel rods could experience peak cladding temperatures above the long-term temperature threshold of 400 °C, reaching up to 461 °C. The peak canister temperature in the simulation is 283 °C. However, the study concluded that the canister would not fail under such conditions. Further, blocked vents or vent channels would not lead to fuel rod failure on a short-term basis.

Unlike a short-term event like a blocked vent channel, a failure in channel design could lead a long-term thermal issue, and the SNF peak cladding temperature could exceed the long-term temperature limit for a duration of multiple years (or decades), although the SNF decay heat is going to decrease with time. Fuel rod failure due to high temperatures alone would not lead to criticality, a release of radioactive material to the environment [for a "leaktight" system, as defined by ANSI N14.5 "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment" (U.S. ANSI 2014)], or a significant increase in radiation exposure of the public or the operation personnel; however, fuel rod failure affects the dry storage system confinement function. In particular, redundancy due to the confinement barrier provided by fuel cladding would not exist if the safety function of the vent channels could not be relied on to preclude the potential for high temperatures that could affect both the integrity of the SNF rods as well as the integrity of the canister.

Based on the discussion summarized above, although thermal components are classified as Category A components in NUREG/CR-6407 (U.S. NRC 1996), the determination of the risk significance for an LAR item that includes a modification of the vent openings is rated as medium.

Gate 1.2.3.1. Outer Cask Shell

NUREG/CR-6407 (U.S. NRC 1996) classified the outer shell as a Category A or B item, depending on the functions the shell needs to fulfill. A controlling factor for the risk significance related to a change in the outer cask shell is the system design characteristic that determines whether or not the outer cask shell provides structural support to the gamma shield (U.S. NRC 1996). In the latter case, a modification of the shell could lead to a rapid loss of gamma shielding during a fire event.

Some dry storage canisters are also licensed for transportation, and the outer shell is important for protecting the confinement barrier (e.g., the welded canister) of the cask when mechanically stressed (e.g., during a transportation accident). Further, the outer cask shell plays an important role during storage. For instance, it protects the shell content from mechanical loads caused by earthquakes or airplane impacts.

Looking at the available NUREG-1864 PRA data (U.S. NRC 2006) a reduction of the protection provided by the outer shell (e.g., due to a design change) could increase the probability of release of radionuclides and noble gas during an earthquake or an airplane impact. However, the maximum possible increase of the overall risk to the public for a 20-year dry storage operation is only 6% (see Table 9) although system redundancy cannot be claimed, since it is expected that an independent confinement barrier such as a welded canister is much weaker than the overpack and could fail as a consequence of overpack failure. Further, a modification of the overpack to a less robust design could make it susceptible to failure due to other accident scenarios not considered in the NUREG-1864 PRA. Thus, the risk significance of a change in the outer cask shell is qualitatively determined as medium.

Table 9. Canister release risk increase assuming an increased canister failure probability due to an improperly modified overpack (Gate 1.2.3.1.).

Phase	Stage	Activity	Event	Initiating Event Frequency	Probability of Release from Canister and Rod	Release Risk
Handling	1	Load SNF in Canister	Fuel Assembly Drop	2.20E-03	6.40E-02	1.41E-04
	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	5	Moving Transfer Cask to Preparation Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	6	Moving Transfer Cask to Preparation Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	7	Moving Transfer Cask to Preparation Area, 3rd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	12	Moving Transfer Cask to Bottom Lid Exchange Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	13	Moving Transfer Cask to Bottom Lid Exchange Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	15	Moving Transfer Cask in Direction Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	18	Lowering Transfer Cask onto Overpack	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
	20	Lifting Canister and Opening Transfer Lid	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
	21	Transferring Canister to Overpack	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase (First Year Only	**2.29E-04**
Transfer	26	Lifting Storage Overpack from Helman Rollers with Overpack Transporter	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	27	Moving Storage Overpack to Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	28	Holding Storage Overpack	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	29	Moving Storage Overpack away from Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	30	Transferring Storage Overpack on Asphalt to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	31	Transferring Storage Overpack on Gravel to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	32	Moving Storage Overpack above Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	33	Lowering Storage Overpack onto Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase (First Year Only)	**0.00E+00**
Storage	34	Dry Storage	Overpack Tip-over due to Seismic Event	7.00E-07	**1.00E+00**	7.00E-07
			Overpack Struck by Aircraft	6.30E-09	**1.00E+00**	6.30E-09
			Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
			Overpack Headed by Aircraft Fuel	3.70E-09	**1.00E+00**	3.70E-09
					Risk Storage Phase (Per Year)	**7.10E-07**
					Total Risk (20 Years)	**2.43E-04**
					Risk Increase Factor	**1.06E+00**

Gate 1.2.3.2. Access Door/Lid Bolts

Some concrete dry storage systems use a shielded access door to close the concrete overpack after being loaded with a canister. The door shields the environment from radiation originating from the canister. The shielded access door is classified a Category B component (i.e., a failure would be a major safety concern) according to NUREG/CR-6407 (U.S. NRC 1996), and the bolts are necessary for holding the shield in place.

Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded. Thus, it is expected that radiation exposure to personnel and the public due to a lack of shielding capability would be detected within a time frame that would allow corrective measures to be put into place. Nonetheless, the design must be reviewed to ensure proper shielding performance.

The risk significance of an LAR item that includes a modification of the access door bolts is evaluated as medium.

Gate 1.2.3.3. Dry Storage Pad

The dry storage pad is the foundation of the dry storage cask system. The cask or concrete overpack rests on the pad. It is classified as a Category C item (minor importance to safety) in NUREG/CR-6407 (U.S. NRC 1996); further, pads are classified as items not important to safety for several approved dry storage systems. The storage pad needs to be able to withstand events like natural hazards, such as earthquakes. Although very unlikely (due to its solid dimensions), the pad could fail in such an event if not properly designed.

In the NUREG-1864 PRA (U.S. NRC 2006), the considered earthquake acceleration that could lead to a cask tip-over is an earthquake causing a ground acceleration of 1.35 g. The corresponding annual event frequency is 7E-7. Although hypothetical, a wrongly designed storage pad could reduce the ground acceleration necessary to tip-over the cask. A more frequent earthquake with a lower ground acceleration could be sufficiently strong enough to lead to such an event. Such an earthquake has a higher frequency than the earthquake considered in NUREG-1864. Considering earthquakes with a lower ground acceleration could increase the risk of confinement failure during a 20-year dry storage period.

In Table 10, the NUREG-1864 PRA was reevaluated with an increased event frequency of a cask tip-over due to an earthquake by a factor of 1,000. This results in a negligible overall risk increase for a 20-year dry storage operation. Thus, the results support the classification of the storage pad as a Category C or not-important-to-safety item, and the risk significance for an LAR item that includes the modification of the storage pad is determined as low.

Table 10. Canister release risk increase assuming an increased canister tip-over frequency due to an improperly modified storage pad (Gate 1.2.3.3.).

Phase	Stage	Activity	Event	Initiating Event Frequency	Probability of Release from Canister and Rod	Release Risk
Handling	1	Load SNF in Canister	Fuel Assembly Drop	2.20E-03	6.40E-02	1.41E-04
	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	5	Moving Transfer Cask to Preparation Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	6	Moving Transfer Cask to Preparation Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	7	Moving Transfer Cask to Preparation Area, 3rd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	12	Moving Transfer Cask to Bottom Lid Exchange Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	13	Moving Transfer Cask to Bottom Lid Exchange Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	15	Moving Transfer Cask in Direction Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
	18	Lowering Transfer Cask onto Overpack	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
	20	Lifting Canister and Opening Transfer Lid	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
	21	Transferring Canister to Overpack	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase (First Year Only	**2.29E-04**
Transfer	26	Lifting Storage Overpack from Helman Rollers with Overpack Transporter	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	27	Moving Storage Overpack to Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	28	Holding Storage Overpack	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	29	Moving Storage Overpack away from Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	30	Transferring Storage Overpack on Asphalt to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	31	Transferring Storage Overpack on Gravel to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	32	Moving Storage Overpack above Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	33	Lowering Storage Overpack onto Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase (First Year Only)	**0.00E+00**
Storage	34	Dry Storage	Overpack Tip-over due to Seismic Event	**7.00E-04**	1.00E-06	7.00E-10
			Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
			Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
			Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (Per Year)	**7.88E-10**
					Total Risk (20 Years)	**2.43E-04**
					Risk Increase Factor	**1.00E+00**

Gate 1.2.3.3. Inner Support Structure

The canister could be stored either horizontally or vertically within the overpack. The inner cask support structure is the structure that holds the inner canister in place. It is classified as a Category B item in NUREG/CR-6407 (U.S. NRC 1996) since a failure of the support structure itself does not directly lead to a loss of confinement. Additionally, a canister breach is required to release its contents.

The PRA presented in NUREG-1864 (U.S. NRC 2006) indicates that, after the canister is welded and leak-tested, a canister drop into the overpack during handling is the event associated with the highest canister release risk (1.57E-5) (see Table 4). This event has the highest probability (2.8E-1) of leading to a canister breach due to the hard impact of the canister without any energy dissipation through a protecting structure, such as impact limiters.

A collapse of the overpack-internal support structure is a similar event, leading to a drop and impact of the unprotected canister within the overpack. Likely, a canister that is horizontally stored in the overpack has a higher chance of breaching than a vertically stored canister due to the expected higher drop height than the vertical configuration. However, the canister impact due the collapse of the internal support structure scenario is expected to be softer compared to a canister impact caused by a drop during transfer with the transfer cask, due to the anticipated lower drop height in the case of a collapsing internal support structure, and the anticipation that some of the kinetic energy would be dissipated by the deformation of the collapsing support structure beneath the canister.

Thus, sufficient safety margins can be assumed, and the risk significance associated with an LAR item that includes a modification of the inner cask support structure is determined as medium.

Gate 1.2.3.5. Cask Hardware

Cask hardware includes keepers, lanyards, small bolts and nuts, cotter pins, etc. Further, the cask hardware is classified as a Category C item in NUREG/CR-6407 (U.S. NRC 1996). It is part of the structure of a dry storage system, but it does not fulfill a specific safety function, and a hardware modification does not significantly affect the confinement, shielding, or criticality control capabilities of the dry storage system.

Thus, the risk significance associated with an LAR item that includes a modification of the cask hardware is determined as low.

Gate 1.2.4.1. Access Door Lifting Lugs

The lifting lugs for the shield access door or lid are operations support items that are needed for a successful opening or closing procedure of the door or lid. A failure could lead to a delay in closure of the cask, and consequently, to an increase of the radiation exposure of the operating personnel. However, the exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. It is expected that radiation exposure to personnel and the public due to a lack of shielding capability would be detected within a time frame that allow putting corrective measures into place.

Thus, the qualitative evaluation of the risk associated with an LAR item that includes modification of the access door lifting lugs is determined as medium, which aligns with the Category B rating of NUREG/CR-6407 (U.S. NRC 1996).

Gate 1.2.4.2. Protective Cover

Some casks designs include a protective cover to further protect the cask from snow and ice loading, and from dust. The item is considered a Category C item according NUREG/CR-6407 (U.S. NRC 1996). However, some applicants have classified the protective cover as an SSC important to safety, and the analyses considers the impact energy absorption of the cover (e.g., in the evaluation of a tornado missile impact). A failure of the protective cover alone is insufficient for a loss of confinement of the system, but a potential loss of protection due to a design modification of the cover needs to be evaluated.

Based on the discussion summarized above, the qualitative evaluation of the risk associated with an LAR item that includes the modification of the protective cover is determined as medium.

Gate 1.2.4.3.1. Outer Shell Lugs/Trunnions/Grapples

Outer shell lugs, trunnions, or grapples refer to the most outer attachments used to lift and move the dry storage cask. Although the term “outer” is used, the attachment point can be located within the outer shell. The risk significance of the LAR item that includes a modification of these components is directly related to the question of whether the outer shell is moved after it was loaded (e.g., within a secondary containment) with SNF, or if the equipment is only used for moving the outer shell or overpack during installation at the storage pad where it resides for long-term storage. If the outer shell is moved after loading, the risk associated with a modification of the lugs, trunnions, or grapples is qualitatively determined as medium, since it could lead to an increase of the drop frequency of the cask, and consequently to an increased probability of a release of radioactive material to the environment, but also considering the large safety margins (an overpack is likely to survive severe accidents, such as an airplane impact). The risk significance of an LAR item that includes a modification of the outer shell lugs, trunnions, or grapples is determined as low when an outer shell or overpack is moved in empty state only.

Gate 1.2.4.3.2. Outer Shell Lugs/Trunnions/Grapples Bolts or Welds

Safe lifting of the outer shell is directly connected to the bolts and welds of the lifting lugs, trunnions, or grapples. It is important to note that the load-bearing joint is not necessarily between the outer shell and the attachment but could rather be located at the interface between an inner component and the attachment. The risk associated with a modification of the bolts or welds is dependent on whether the shell is lifted only in empty state, or also after loading. If the outer shell is moved after loading, the risk associated with a modification of the lug, trunnion, or grapple bolts or welds is qualitatively determined as medium, since it could lead to an increase of the drop frequency of the cask, and consequently to an increased probability of a release of radioactive material to the environment, but also considering the large safety margins (an

overpack is likely to survive severe accidents, such as an airplane impact). The risk significance of an LAR item that includes a modification of the outer shell lugs, trunnions, or grapples bolts or welds is determined as low when an outer shell or overpack is moved in empty state only.

Gate 1.2.4.4. Security Lockwire and Seals

Security lockwires and seals are classified as Category C items in NUREG/CR-6407 (U.S. NRC 1996). They are put into place to indicate if a cask has been subjected to unauthorized tampering, but they do not fulfill a function that is directly related to safety. Thus, the risk associated with a modification of these items is qualitatively determined as low.

Gate 1.2.4.5. Shielding Shell

Some metal casks use a thin shell as the most outer layer to cover the shielding components and to protect more sensitive parts of the outer shell from weather exposure. However, this shell does not fulfill any safety function, and is classified as Category C items in NUREG/CR-6407 (U.S. NRC 1996). Thus, the risk associated with a modification of the shielding shell is qualitatively determined as low.

Gate 1.2.5. Lightning Protection

Some dry storage systems are grounded to protect it from lightning strikes. Lightning protection system components are not classified in NUREG/CR-6407 (U.S. NRC 1996), but some dry storage designs classify lightning protection as an SSC important to safety.

In NUREG-1864 (U.S. NRC 2006) the effects of a lightning strike on an ungrounded dry storage system were evaluated. The analyses indicate that the effects are marginal, and that a direct lightning strike in an ungrounded system does not pose a threat to the confinement capabilities. A report (Yugo 2002) developed by researchers of the Oak Ridge National Laboratory (ORNL) for the U.S. NRC came to similar conclusions.

Thus, the risk significance associated with an LAR item that includes a modification of the lightning protection system is qualitatively determined as low, related to the confinement, criticality, and shielding of the system. However, an improperly functioning lightning protection system could increase the chance of workers nearby being struck by lightning.

2.5.3 Design – Transfer Cask

Gate 1.3.1.1. Water Jacket

The transfer cask including its components, are classified as a Category B item in NUREG/CR-6407 (U.S. NRC 1996).

Some transfer casks have a water jacket that provides neutron shielding. A modification of this jacket could increase the potential radiation exposure of operating personnel. However, during SNF handling, the radiation level is usually measured, and the radiation exposure of personnel is recorded by personal dosimeters. Insufficient shielding of the transfer cask would

likely be detected. Nevertheless, there is a risk of rapid water loss by the jacket, in case the design is flawed, exposing personnel to a high radiation immediately.

Based on the discussion summarized above, the risk significance of an LAR item that includes a modification of the water jacket is estimated as high.

Gate 1.3.1.2. Lead Shield

The transfer cask, including its components, are classified as a Category B item in NUREG/CR-6407 (U.S. NRC 1996).

Transfer casks typically have a lead shield that provides protection from gamma radiation. A modification of this shield could increase the potential radiation exposure of operating personnel. However, during SNF handling, the radiation level is usually measured, and the radiation. exposure of personnel is recorded by personal dosimeters. Insufficient shielding of the transfer cask would likely be detected.

Furthermore, the transfer cask, to a certain degree, provides physical protection to the canister. An improper modification of the shell of the transfer cask, including the lead shield, could change the response of the canister to mechanical loads and increase the risk of confinement failure in case the loaded transfer cask is dropped during the handling phase. A highly conservative evaluation of the potential risk increase due to such modifications includes the assumption of an increased likelihood of a canister or canister weld breach during transfer cask movements. A reevaluation of the NUREG-1864 PRA (U.S. NRC 2006) considering this assumption indicates an increase of the total release risk by a factor of 4 (see Figure 9).

Based on the considerations regarding gamma shielding and protection of the canister from impact loads by the transfer cask shell, the risk significance of an LAR item that includes a modification of the gamma shield is estimated as medium.

Table 11. Canister release risk increase assuming an increased canister or canister weld breach probability due to an improperly modified transfer cask lead shield (Gate 1.3.1.2.).

Phase	Stage	Activity	Event	Initiating Event Frequency	Probability of Release from Canister and Rod	Release Risk
Handling	1	Load SNF in Canister	Fuel Assembly Drop	2.20E-03	6.40E-02	1.41E-04
	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	5	Moving Transfer Cask to Preparation Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	6	Moving Transfer Cask to Preparation Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	7	Moving Transfer Cask to Preparation Area, 3rd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	12	Moving Transfer Cask to Bottom Lid Exchange Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	13	Moving Transfer Cask to Bottom Lid Exchange Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	15	Moving Transfer Cask in Direction Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
	18	Lowering Transfer Cask onto Overpack	Dropping Transfer Cask, Height 100 ft	5.60E-05	1.00E+00	5.60E-05
	20	Lifting Canister and Opening Transfer Lid	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
	21	Transferring Canister to Overpack	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase (First Year Only	**9.56E-04**
Transfer	26	Lifting Storage Overpack from Helman Rollers with Overpack Transporter	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	27	Moving Storage Overpack to Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	28	Holding Storage Overpack	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	29	Moving Storage Overpack away from Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	30	Transferring Storage Overpack on Asphalt to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	31	Transferring Storage Overpack on Gravel to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	32	Moving Storage Overpack above Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	33	Lowering Storage Overpack onto Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase (First Year Only)	**0.00E+00**
Storage	34	Dry Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
			Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
			Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
			Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (Per Year)	**8.89E-11**
					Total Risk (20 Years)	**9.56E-04**
					Risk Increase Factor	**4.17E+00**

Gate 1.3.1.3. Top Lid

The transfer cask including its components, are classified as a Category B items in NUREG/CR-6407 (U.S. NRC 1996).

Transfer casks typically have a top lid that provides protection from radiation. A modification of this lid could increase the potential radiation exposure of the operating personnel. However, during SNF handling, the radiation level is usually measured, and the radiation exposure of personnel is recorded by personal dosimeters. Insufficient shielding of the transfer cask would likely be detected. Furthermore, the transfer cask provides, to a certain degree, physical protection to the canister. Therefore, a modification of the shell of the transfer cask, including the top lid, could increase the risk of a confinement failure in case the loaded transfer cask is dropped during handling. The factor of potential release risk increase is similar to the factor computed for a modification of the lead shield, which is approximately 4 (see Table 11 and the corresponding Gate 1.3.1.2.).

Based on the considerations regarding shielding and protection of the canister from impact loads by the transfer cask shell, the risk significance of an LAR item that includes a modification of the transfer cask top lid is estimated as medium.

Gate 1.3.1.4. Bottom Lid

The risk significance of a modification of the design of the bottom lid of the transfer cask (in this rationale, the term “bottom lid” includes the transfer cask pool lid or the canister-to-overpack transfer cask device) depends on whether or not the SNF handling operations or transfer phase include a situation in which the loaded canister rests on the bottom lid, while the loaded canister is lifted or rests on the storage overpack. Such a situation would allow a drop and subsequent impact of an unprotected canister if a malfunction of the bottom lid would occur. For instance, modifying the canister-to-overpack transfer cask device design could increase the probability of a canister drop during transfer to the overpack, and consequently, increase the canister release risk during a 20-year dry storage operation by a factor larger than 2,000 (see Table 8 including the corresponding rationale of Gate 1.1.5.1.1.). Note that generally, a drop of the unprotected canister due to a malfunction of the correctly designed bottom lid is unlikely.

If the canister does not rest unsecured on the bottom lid during any phase of the operations, the lid’s main function is to provide radiological protection and structural protection during a transfer cask drop. However, a modification of the shell of the transfer cask, including the bottom lid, could increase the likelihood of a confinement failure in case the loaded transfer cask is dropped during handling. The factor of potential release risk increase is similar to the factor computed for an improper modification of the lead shield, which is approximately 4 (see Table 11 including the corresponding rationale of Gate 1.3.1.2.).

Based on the discussion summarized above, the risk significance of an LAR item that includes a design modification of the transfer cask bottom lid (i.e., the pool lid or the canister-to-overpack transfer device) is dependent on the cask operational procedures. It is estimated as high if the procedures include situations in which the canister rests unsecured on the bottom lid while the transfer cask is lifted, because such a situation could lead to a canister drop in case of a lid structural failure. If a canister drops due to transfer cask bottom lid failures can be excluded, the risk significance is estimated as medium.

Gate 1.3.2.1. Transfer Cask Shell

The transfer cask (including its components) is classified as a Category B item in NUREG/CR-6407 (U.S. NRC 1996).

The transfer cask shell includes all its outermost components except the water jacket, lids, and shields. The shell components provide physical protection to the canister. Therefore, any modification of the transfer cask shell could increase the risk of a confinement failure in case the loaded transfer cask is dropped during handling. The factor of potential release risk increase is similar to the factor computed for a modification of the lead shield, which is approximately 4 (see Table 11 including the corresponding rationale of Gate 1.3.1.2.).

Based on the discussion summarized above, the risk significance of an LAR item that includes a modification of the transfer cask shell is estimated as medium.

Gate 1.3.2.2. Transfer Cask Trunnions

Modifying the transfer cask trunnions could affect the probability of accidental drops of the loaded cask during SNF transfer. A reevaluation of NUREG-1864 PRA (U.S. NRC 2006) indicates a potential risk increase by a factor above 4,000 (see Table 12). This significant increase in risk can be traced back to the lower protective capability of the transfer cask compared to more robust systems, such as the overpack.

Thus, the risk significance of an LAR item that includes a modification of the transfer cask trunnions is estimated as high.

Table 12. Canister release risk increase assuming an increased transfer cask frequency due to improperly modified transfer cask trunnions (Gate 1.3.2.2.).

Phase	Stage	Activity	Event	Initiating Event Frequency	Probability of Release from Canister and Rod	Release Risk
Handling	1	Load SNF in Canister	Fuel Assembly Drop	2.20E-03	6.40E-02	1.41E-04
	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	1.00E+00	1.00E+00	1.00E+00
	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	5	Moving Transfer Cask to Preparation Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	6	Moving Transfer Cask to Preparation Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	7	Moving Transfer Cask to Preparation Area, 3rd Section	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	12	Moving Transfer Cask to Bottom Lid Exchange Area, 1st Section	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	13	Moving Transfer Cask to Bottom Lid Exchange Area, 2nd Section	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	15	Moving Transfer Cask in Direction Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
	18	Lowering Transfer Cask onto Overpack	Dropping Transfer Cask, Height 100 ft	1.00E+00	2.00E-02	2.00E-02
	20	Lifting Canister and Opening Transfer Lid	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
	21	Transferring Canister to Overpack	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase (First Year Only	**1.02E+00**
Transfer	26	Lifting Storage Overpack from Helman Rollers with Overpack Transporter	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	27	Moving Storage Overpack to Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	28	Holding Storage Overpack	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	29	Moving Storage Overpack away from Preparation Area	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	30	Transferring Storage Overpack on Asphalt to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	31	Transferring Storage Overpack on Gravel to Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	32	Moving Storage Overpack above Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
	33	Lowering Storage Overpack onto Storage Pad	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase (First Year Only)	**0.00E+00**
Storage	34	Dry Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
			Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
			Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
			Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (Per Year)	**8.89E-11**
					Total Risk (20 Years)	**1.02E+00**
					Risk Increase Factor	**4.45E+03**

Gate 1.3.2.3. Transfer Cask Hardware

The transfer cask hardware includes keepers, bolts, nuts, cotter pins, detent pins, lockwires, or lanyards that prevent bolt and nuts from loosening or being misplaced. Hardware parts are classified as Category C components according to NUREG/CR-6407 (U.S. NRC 1996).

The function of the hardware is limited to structural integrity (U.S. NRC 1996). A modification of the canister hardware does not have an effect on the core safety functions of the system (i.e., shielding, confinement, or criticality control). Thus, the qualitative determination for the risk significance of canister hardware modification is low.

2.5.4 Approved Content

Gate 2.1.1. Increased Burnup

The SNF burnup affects the reactivity, radioactivity, heat load, and fuel cladding condition of the assemblies. Typically, the reactivity of a fuel assembly decreases with increasing burnup, however, fuel assemblies that contain burnable absorbers can show a more complex criticality-burnup relationship. Nevertheless, the reactivity of fresh fuel is considered bounding in a criticality safety analysis (U.S. NRC 2002, U.S. NRC 2012b) Thus, increasing the permitted burnup of the assemblies in a SNF dry storage system should reduce the reactivity of the canister content. Special care needs to be taken in case the burnup was credited in original criticality evaluations considering the original SNF assembly burnups.

Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded. Thus, it is expected radiation exposure to personnel and the public due to an increased SNF assembly burnup would be detected within a time frame that allows putting corrective measures into place. Nonetheless, a review should ensure the system would meet dose limits.

The heat load of the SNF assemblies typically increases with higher burnup. A higher heat load could lead to SNF degradation effects due to an increase in peak cladding temperatures of the fuel rods. However, the regulatory cladding temperature thresholds do not represent hard limits indicating cladding failure or lack thereof. Rather, these temperature thresholds were put into place to prevent cladding from deforming or degrading due to effects like hydride embrittlement or hydride reorientation; recent research (Eidelpes, Ibarra and Medina 2019), however, indicates latter effects might not play as important a role as previously thought.

Although it is important to recognize that a release [from a "leaktight" system, as defined by ANSI N14.5 "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment" (U.S. ANSI 2014)] due to an increased SNF burnup requires a failure of both SNF and confinement barrier, redundancy cannot be claimed since the increased heat load due to the increased burnup could challenge both items.

Based on the discussion above, the risk significance of an LAR item that includes an increase in approved SNF burnup is qualitatively evaluated as medium.

Gate 2.1.2. Burnup Credit

Crediting the SNF burnup in the criticality evaluations reduce the overall safety margin to criticality. But, related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010). Nevertheless, crediting the burnup requires the consideration of a more complex parameter constellation in the criticality evaluations. Special care is required to ensure criticality safety during very long (more than 100 years) term dry storage, since the reactivity could increase due to decay of ^{240}Pu.

Based on the discussion above, the risk significance of crediting the SNF burnup is qualitatively evaluated as medium.

Gate 2.2. Increased Enrichment

The ^{235}U enrichment in current light water reactor fuel assemblies reaches up to 5%.

Changing the fuel enrichment affects the reactivity of the SNF in a dry storage cask, as well as the shielding requirements of the cask and radionuclide available for release. An increased enrichment of the approved canister content also may require an update of the criticality safety evaluations and the shielding evaluations. Further, storing SNF with a significantly different enrichment would require the evaluation of the availability of appropriate criticality benchmarks. Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded. Thus, it is expected that radiation exposure to personnel and the public due to an increased fuel enrichment would be detected within a time frame that allows corrective measures to be put into place. Nonetheless, a review should ensure the system would meet dose limits.

Based on the discussion above, the risk significance of an LAR item that includes an increased enrichment of the approved canister content is qualitatively evaluated as medium.

Gate 2.3. Uranium Mass

The amount of fissile material in the dry storage system affects criticality safety and shielding requirements of the dry storage system. Increasing the uranium mass likely increases the reactivity of the dry storage cask content. Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded. Thus, it is expected that radiation exposure to personnel and the public due to an increased uranium mass would be detected within a time frame that allows corrective measures to be put into place. Nonetheless, a review should ensure the system would meet dose limits.

Based on the discussion above, the risk significance of an LAR item that includes modification of the permitted uranium mass per SNF assembly is qualitatively evaluated as medium.

Gate 2.4.1. Lattice Type

The use of a different lattice type of SNF assemblies modifies the configuration of the fissile material within the cask. Thus, the criticality safety needs to be revaluated. Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

Based on the discussion above, the risk significance of an LAR item that includes a modification of the permitted SNF lattice type is qualitatively evaluated as medium.

Gate 2.4.2. Different Geometry

The inclusion of a different SNF assembly geometry modifies the configuration of the fissile material within the cask. Thus, the criticality safety needs to be revaluated. Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly

reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

Based on the discussion above, the risk significance of an LAR item that includes a modification of the permitted SNF assembly geometry is qualitatively evaluated as medium.

Gate 2.4.3. Assembly Mass

The assembly mass could potentially affect the structural integrity of the dry storage cask. A small increase or a decrease in mass of the SNF assemblies likely has no significant effect on the structural response or safety of the system. A significant increase of the assembly mass, however, could lead to a different response of the assemblies, the basket, the canister and/or the cask/overpack, including supporting structure. Potential consequences could be a loss of structural integrity of the fuel assemblies, a displacement neutron poisons, or a basket failure, which could increase the reactivity in the cask in case a moderator is present. In some instances, additional structural reviews and assessments may need to be performed, to conclude on a risk significance.

Based on the discussion above, the risk significance of an LAR item that includes a modification of the permitted SNF assembly weight is qualitatively evaluated as low for small changes in the permitted assembly mass, and medium (or could be even high) for significant changes in the assembly mass.

Gate 2.5.1. Damaged Fuel

The current definition of damaged SNF is a performance-based definition (i.e., damaged SNF include fuel rods or fuel assemblies which cannot perform their intended functions) (U.S. NRC 2007). Further, 10 CFR 72.122 (h) requires that, during storage, SNF must be protected against degradation that leads to SNF gross rupture, but assemblies with dents in rods, bent or missing structural members, small cracks, or missing rods are typically not considered damaged.

Usually, damaged SNF is canned before placement in a SNF cask to confine gross fuel particles, debris, or damaged assemblies to a known volume within a cask. This process is necessary to comply with criticality, thermal, shielding, and structural requirements on SNF, and to permit handling and retrievability (U.S. NRC 2007). Nevertheless, even if fuel particles of the damaged SNF are released into the cask, the effects on safety of the dry storage system are insignificant. An accumulation of fissile material at the bottom of the canister or inner cask could affect the reactivity of the dry storage cask content and could locally impact canister-internal temperatures. Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

Further, the shielding capability of a dry storage system is usually designed for an intact SNF configuration. Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the regulatory dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask

is loaded. Thus, it is expected that radiation exposure to personnel and the public due to a lack of shielding capability would be detected within a time frame that allows corrective measures to be put into place. Nonetheless, a review should ensure the system would meet dose limits.

Although the fuel cladding is the inner-most confinement barrier in a dry storage cask, the confinement capability of the canister or inner cask [for a "leaktight" system, as defined by ANSI N14.5 "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment" (U.S. ANSI 2014)] is adequate to prevent a release of radioactive material or noble gases to the environment.

Based on the discussion summarized above, an LAR item that includes the storage of damaged fuel in a dry storage system requires a detailed review of the updated safety evaluations, but the risk significance of such a modification can be qualitatively evaluated as medium.

Gate 2.5.2. Cooling Time

The amount of time the SNF assemblies are placed in a wet storage pool affects the heat load in a canister, and to a certain degree, their radioactivity of the SNF assemblies. The longer the cooling time, the more heat can decay. If the maximum heat load in a dry storage cask is increased, the permitted peak cladding temperature limits of some fuel rods may be exceeded. In an extreme case, this could lead to SNF cladding degradation. However, fuel cladding failure caused solely by high temperatures is unlikely. Further, fuel cladding failure does not lead to a direct release of canister content to the environment, although redundancy cannot be claimed because the increased heat load would also challenge the confinement barrier. Furthermore, additional care needs to be taken for special cases such as non-leak tight systems.

Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Detailed surface dose rate measurements are taken when a dry storage cask is loaded, and it is expected that radiation exposure to personnel and the public due to a reduced cooling time would be detected within a time frame that allows corrective measures to be put into place. Nonetheless, a review should ensure the system would meet dose limits.

Based on the discussion summarized above, an LAR item that includes a reduction of the cooling time of SNF assemblies requires a thermal reevaluation of the storage cask. The risk significance of such a modification is medium.

Gate 2.5.3. Heat Load

The heat load in a SNF dry storage cask is an important parameter. An increase of the total heat load would require a detailed thermal review. For instance, an increased cladding temperature affects the confinement barrier integrity, because the permitted peak cladding

temperature limits of some fuel rods may be exceeded. In an extreme case, this could lead to SNF cladding degradation. However, in a "leaktight" system [as defined by ANSI N14.5 "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment" (U.S. ANSI 2014)], fuel cladding failure due to an increased heat load does not lead to a direct release of canister content to the environment. However, redundancy cannot be claimed, because the increased heat load increases the thermal challenge to the canister's confinement safety function.

Based on the discussion summarized above, an LAR item that includes an increased heat load of SNF assemblies require a thermal reevaluation of the storage cask, but the risk significance of such a modification is medium.

Gate 2.6. IFBA Rods

The reactivity of Integral Fuel Burnable Absorber (IFBA) rods changes with increasing burnup, which needs to be considered in the criticality safety evaluations when crediting burnup. This change is partly associated with the depletion of burnable absorbers within the rods during in-reactor operation. Adding undecayed burnable absorbers to the approved canister content likely reduces the reactivity of the content in the dry storage system, and removing burnable absorbers increases the reactivity.

Criticality safety in a dry storage system, is typically evaluated assuming fresh fuel and disregarding the presence of burnable absorbers. This approach results in an upper-limit, bounding criticality safety evaluation with conservative margins (U.S. NRC 2002). More recently, cask licensees have tended to credit the burnup of SNF when evaluating the criticality safety of dry storage cask contents. Such an approach needs to consider the decay process of burnable absorbers on the reactivity of the fissile material in the cask, as outlined in the U.S. NRC Interim Staff Guide (ISG) 8, Revision 3. Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

Based on the discussion as outlined above, modifying, removing, or introducing burnable absorbers such as IFBA rods into the canister could require a more complex reevaluation of the criticality safety of the dry storage system, specifically if the rod burnup is credited. The risk significance of such a modification is qualitatively evaluated as medium.

Gate 2.7. Cask Loading Pattern

The cask loading pattern of the SNF influences the fuel assembly temperatures, radiation levels, and criticality in a dry storage system. In an extreme case, a modification of the pattern could lead to fuel cladding degradation and consequent cladding failure. However, the effect of the cask loading pattern on the SNF peak cladding temperature is likely smaller than a reduction in SNF assembly cooling time (see rationale of Gate 2.5.2.), or an increase in total cask heat load (see rationale of Gate 2.5.3.).

Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that the original criticality safety evaluations normally do not credit the potential of

criticality-reducing effects of cask loading patterns (such as a placement of more reactive SNF assemblies in the outer canister regions where the neutron leakage is more pronounced). Thus, a modification of the cask loading pattern should not increase the risk of reaching criticality.

Further, to reach criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

The locations of the fuel assemblies within the cask affect the radiation levels in proximity to the cask, although the effect is likely small. Dose limits to the public and to the operating personnel of a dry storage system are regulated under 10 CFR 20 and 10 CFR 72. However, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary). Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage system is loaded. Thus, it is expected that radiation exposure to personnel and the public due to an increased fuel enrichment would be detected within a time frame that allows for putting corrective measures into place. Nonetheless, a review should ensure the system would meet dose limits.

The qualitative assessment of the cask loading pattern considered cladding failure due to a thermal issue as the consequence with the highest chance of occurrence. However, such an event would not cause an immediate failure of any of the main safety functions of the dry storage system, although there is a potential for increased release due to cladding failure for non-leaktight confinement. For a release of radioactive material to the environment from a “leaktight” system [as defined by ANSI N14.5 “American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment” (U.S. ANSI 2014)], the confinement barrier (welded canister, or cask seal) would need to fail in addition to the cladding. A reconfiguration of the assembly loading pattern should not increase the total SNF heat load within the cask, and thus, should not significantly challenge the confinement barrier.

Gate 2.9. Cladding Type

Some of the typical cladding types used in U.S. nuclear power plants are Zircaloy-2, Zircaloy-4, M5, ZIRLO, and optimized ZIRLO. Differences in in-reactor corrosion behavior and creep behavior during long-term storage for fuel rods with different cladding alloy types can be observed (Eidelpes, Ibarra and Medina 2019, Spilker, et al. 1997). However, the commercially available cladding alloys proved to be reliable materials for SNF dry storage. The following discussion focus on these currently approved cladding materials and does not specifically consider new materials under development, such as those for accident tolerant fuels.

In case the cladding fails to provide structural support to the fuel, no significant effect on the safety of the dry storage system is expected, because the inner cask or canister shell of a “leaktight” system [as defined by ANSI N14.5 “American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment” (U.S. ANSI 2014)] provides sufficient confinement of the radionuclides and noble gases released from the rods. However, the

magnitude of a potential release is affected significantly by the extent the fuel cladding is damaged.

Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to note that criticality in a dry storage cask requires a moderated system (Alsaed 2018), and likely, significantly reconfigured SNF. Further, criticality safety margins of current systems are typically large (K. Banerjee 2015, Rigby 2010).

It is important to note that NRC regulations require that fuel cladding be protected against degradation or otherwise confined such that fuel does not pose operational safety problems when the fuel is removed from storage (e.g., a release during operations when the cask or canister is not providing confinement). Also, it is important to recognize that the cladding is an important component in the multi-barrier system of a SNF dry storage cask. Further, additional care needs to be taken for special cases such as non-leak tight systems (e.g., the TMI casks at INL). Considering the relatively low risks associated with normal storage operations using the currently approved cladding materials, but also factoring unique risks associated other instances where a "leaktight" boundary [as defined by ANSI N14.5 "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment" (U.S. ANSI 2014)] may not be present, the risk significance of a change to cladding type is evaluated as medium.

Gate 2.9. Reactor Type

Boiling water reactor (BWR) SNF assemblies are smaller and lighter than pressurized water reactor (PWR) assemblies. Typically, more BWR than PWR assemblies can be stored in a SNF dry storage system. Changing the approved content or adding another assembly type requires a reevaluation of the structural integrity of the cask and the supporting structure, a reevaluation of the criticality safety, a thermal reevaluation, and a reevaluation of the shielding capabilities of the system.

Insufficient shielding is likely detected before significant consequences occur. The exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks and the distance of the storage systems to the installation boundary). A facility operator is required to ensure compliance with the dose limits. The radiation exposure to loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded. Thus, it is expected that radiation exposure to personnel and the public would be detected within a time frame that allows for putting corrective measures into place. Nonetheless, a review should ensure the system would meet dose limits.

Thus, the risk significance of an LAR item that includes the storage of SNF assemblies from another reactor type is estimated as medium.

Gate 2.10. Non-Fuel Hardware

Non-fuel hardware includes control components, neutron poison inserts, guide tube hardware, axial power shaping rods, or thimble plug devices ("spiders"). Adding non-fuel hardware to the approved dry storage system content can increase the content mass significantly. For instance, estimates for non-fuel-bearing components range from 25 to 50 kg per assembly,

excluding cruciform-shaped control blades of certain BWR assembly cells (Luksic, et al. 1986). Thus, adding significant amounts of non-fuel hardware to the approved dry storage canister content requires a reevaluation of the structural integrity of the canister.

Further, irradiated non-fuel hardware could increases the total heat load of the canister or affect the temperature decay of the SNF assemblies, although the effect is likely small. Similarly, additional irradiated material in the dry storage system could increase the radioactivity. Both effects require reevaluation of the system by a reviewer. Nevertheless, insufficient shielding is likely going to be detected before significant consequences occur. For instance, the exposure of the public to radiation originating from a dry storage system is dependent on numerous factors (e.g., the number of casks or the distance from the storage system to the installation boundary).

Typically, the safety margins are relatively large. A facility operator is required to ensure compliance with the dose limits. The exposure to radiation of loading and operating personnel is actively monitored, and typically recorded with personal radiation dosimeters. Further, detailed surface dose rate measurements are taken when a dry storage cask is loaded, therefore, it is expected that radiation exposure to personnel and the public due to inclusion of non-fuel hardware would be detected within a time frame that allows corrective measures to be put into place. Nonetheless, a review should ensure the system would meet dose limits.

Based on the rationale summarized above, the risk significance of an LAR item that includes the addition of non-fuel hardware to the canister content is qualitatively determined as medium.

2.5.5 Evaluation

Gate 3.1. Method

A deviation from the methodologies used to conduct the safety evaluations summarized in the FSAR poses a significant risk due to the complexity of the review of such deviation. Thus, the risk significance of an LAR item that includes a deviation from the original evaluation method is qualitatively evaluated as high.

Gate 3.2. Code/Standard

A deviation from the codes and standards previously used in the FSAR using a different, approved code or standard requires an evaluation by a reviewer. The new code or standard needs to be evaluated regarding applicability. The risk significance of an LAR item that includes such a deviation is evaluated as medium.

2.5.6 Editorial Changes

Gate 4.1. Typographical Errors

The risk significance of an LAR item that includes the correction of typographical errors in the FSAR, TSs, or CoC is qualitatively evaluated as low since such a change likely does not directly affect current or future dry storage operating procedures.

Gate 4.2.1. FSAR

The risk significance of an LAR item that includes the revision of definitions and/or provision of clarifications in the FSAR is qualitatively evaluated as medium since such a change could affect current or future dry storage operating procedures.

Gate 4.2.2. Technical Specifications

The technical specifications define the conditions that have been deemed necessary for safe fabrication and operation of dry storage systems. Specifically, they define operating limits and controls, monitoring instruments and control settings, surveillance requirements, design features, and administrative controls and programs that ensure safe operation. Consequently, any change in the TSs requires a careful review.

The risk significance of an LAR item that includes the revision of definitions and/or provision of clarifications in the TSs is qualitatively evaluated as high.

Gate 4.2.3. CoC

The risk significance of an LAR item that includes the revision of definitions and/or provision of clarifications in the license or CoC is qualitatively evaluated as high since such a change could directly affect current dry storage system design, fabrication, and operating procedures that were previously determined by the NRC to define the key aspects of the storage system's safety basis. The impact of the change must be evaluated because of the significance of the CoC document.

2.5.7 Additional Gates

Gate 9. Unevaluated Items

For LAR items that are not evaluated in this document, the reader is encouraged to determine the risk significance by evaluating the request according to the criteria listed in Table 1. If unclear, conservatively, a high risk significance should be assumed.

3. CONCLUSION

The combination of the rationale document and the developed tree diagram should provide the user with the information required to assess requested changes to a dry storage system. The documents should improve the efficiencies in the review process as it makes LAR reviews more consistent in terms of resource allocation, depth, and breadth of the review. For those LAR items or dry storage system-specific characteristics not discussed in the rationales, the user is encouraged to assess whether the provided risk estimates are adequate or if additional precautions should be taken.

4. FUTURE OPPORTUNITIES

There are number of ways to improve the usefulness of this risk tool by extending its applicability and improving its precision with more data. For instance, the completion of the next two tasks of the project, which include the incorporation of risk insights associated with LAR for SNF transportation casks and other regulatory actions could extend the applicability of the tool. Building user experience will help identifying additional opportunities for improvement, especially on the workflow when using this tool. Finally, the research activity that could lead to the most significant improvements of the tool would be the construction of additional SNF dry storage system, and SNF transportation system PRAs, along with the gathering of more data on failures of the corresponding components and failures in operator actions and procedures associated with SNF loading, dry storage and transportation. This could lead to a less conservative approach when assessing the risks inherent to dry storage or transportation system LARs while maintaining a focus on safety. Probabilistic risk assessments could be designed for all portions of the used fuel cycle including wet storage, cask loading, and onsite dry cask storage, as well as transportation of the dry casks, to fully assess the areas of elevated risk to the general public.

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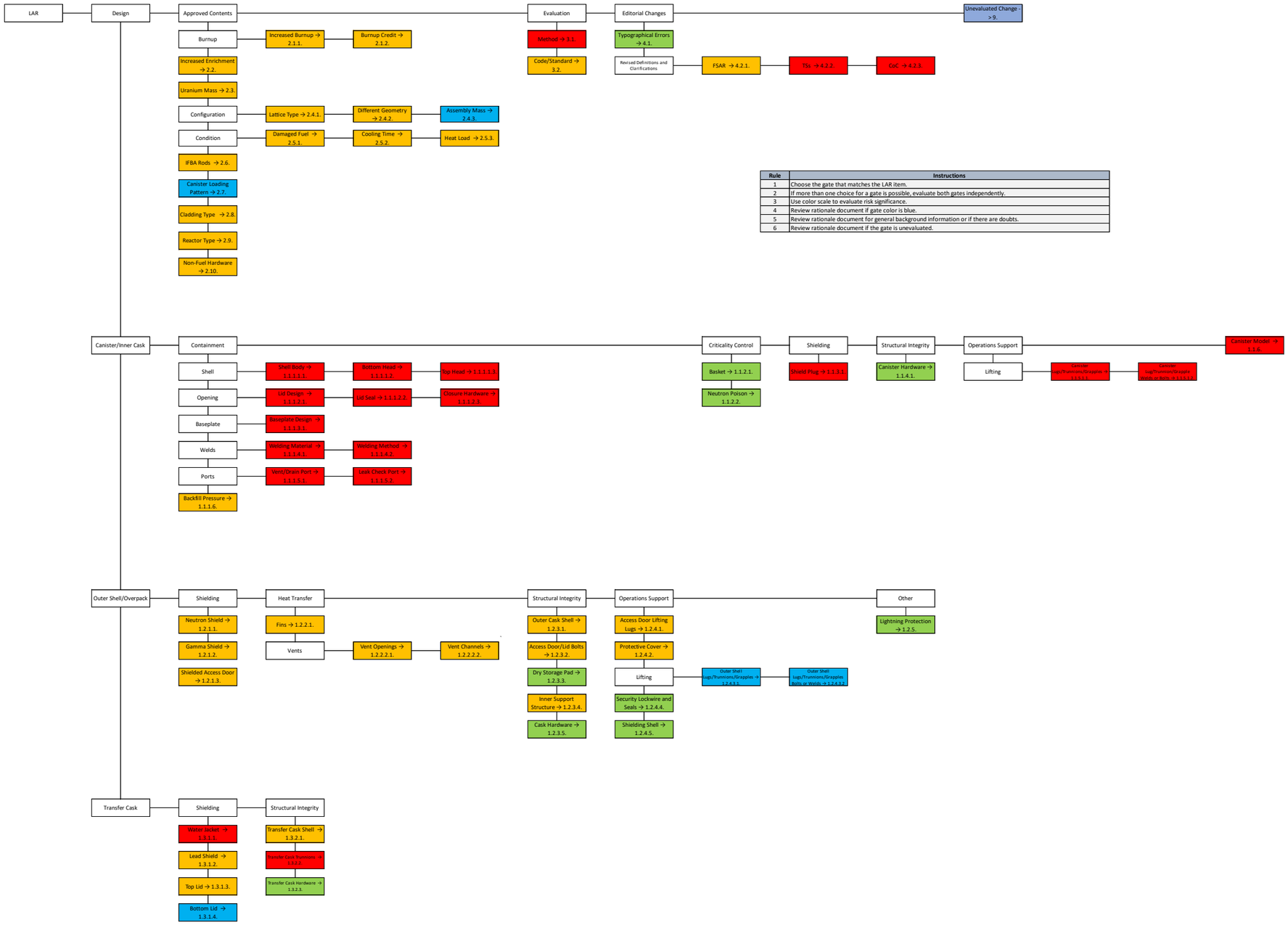
APPENDIX

The tree diagram of the developed LAR dry cask risk tool is presented in Figure A.1.

Criterion: Risk Significance	A: Factor of Risk Increase*	B: Effect on current/future operations	C: Redundancy	D: Safety margins	E: Evaluation Complexity	F: Flaw/Error Detection Probability	LAR Review Recommendations
Low	< 2 or a Decrease in Risk	Yes or No	Exists	Small to Large	Simple	High	Efficient
Medium	≥ 2 and < 10	Yes	Nonexistent	Large	Simple to Complex	Medium to High	In Detail
High	≥ 10	Yes	Nonexistent	Small	Simple to Complex	Low to High	Extensive, Thorough, Very Detailed
See Rationale	No risk significance estimation or review recommendations possible without consideration of additional factors.						

*This factor describes the increase in canister release risk due to an LAR item, compared to the baseline release risk calculated for a dry storage operation described in NUREG-1864. This criterion is only applicable if a quantitative risk sensitivity study is possible.



Rule	Instructions
1	Choose the gate that matches the LAR item.
2	If more than one choice for a gate is possible, evaluate both gates independently.
3	Use color scale to evaluate risk significance.
4	Review rationale document if gate color is blue.
5	Review rationale document for general background information or if there are doubts.
6	Review rationale document if the gate is unevaluated.

Figure A.1. Tree diagram of risk tool.