INL/EXT-20-58896
Revision 1

Development of Dry Cask Risk Tools

July 2022

Technical Report

Elmar Eidelpes, John Biersdorf, Robby Christian, and Kurt Vedros

Idaho National Laboratory

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ACRONYMS

BWR	boiling-water reactor
CFR	Code of Federal Regulations
CoC	certificate of compliance
EPRI	Electric Power Research Institute
FSAR	final safety analysis report
HI-STORM	Holtec International Storage Module
HI TRAC	Holtec International Transfer Cask
INL	Idaho National Laboratory
ISFSI	independent spent fuel storage installation
LAR	license amendment request
MPC	multipurpose canister
NRC	Nuclear Regulatory Commission
NWTRB	Nuclear Waste Technical Review Board
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
SER	safety evaluation report
SSC	structures, systems, and components
SNF	spent nuclear fuel
TMI	Three Mile Island
TS	technical specifications

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1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has often expressed a desire to consider information on risk in its decision-making processes. The Probabilistic Risk Assessment Policy Statement, published in 1995, formalized the commission's commitment to risk-informed regulation through the expanded use of probabilistic risk assessments (PRAs). While a great deal of work has been done to incorporate risk insights into the regulatory framework for at-power nuclear reactors, less progress has been made to incorporate risk insights into the spent nuclear fuel (SNF) dry casks and nuclear waste transportation areas of the nuclear fuel cycle.

The NRC has contracted with Idaho National Laboratory (INL) to develop a tool that incorporates information from dry cask 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: Develop a tool that supports the evaluation of risk related to license amendment requests (LAR) for SNF dry cask storage systems

Task 2: Expand the tool to include SNF transportation systems in future work

Task 3: Address eventual additional regulatory applications (e.g., oversight) for dry cask storage or transportation in future work.

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 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 identifying and prioritizing LAR reviews related to SNF dry storage applications. Fundamentally, the methodology should support the NRC resource allocation strategy by providing a basic, risk-informed framework that can be used to define recommendations of the depth and breadth of LAR reviews. The tool will use quantitative, qualitative, and semiquantitative 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 the NRC, is a tree diagram, 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 effective NRC review process 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 specific LAR item or dry storage system details. It is imperative to recognize that the capability of this tool is limited to provide guidance. Further, the user of this tool must be aware that the risk posed by an LAR item to the safety of the dry storage system could be significantly higher (or lower) than we have outlined in this document due to specific details or

combinations of multiple individual LAR items that cannot be accurately represented without conducting in-depth assessments.

2 METHODOLOGY

2.1 Tool Structure

The general approach to tackling the first task of this project was to develop a tree diagram (see “Dry Storage Risk Tool.xlsx” and Appendix of this document) that provides the user with preliminary estimations of the risk significance of LAR items. 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. The tree diagram is accompanied by a rationale document (i.e., this present report) to establish the rationales and criteria that were used to estimate the risk significances of a number of potential changes. Dry Storage Risk Tool users are encouraged to study the applicable rationale so they can assess whether the estimated risk is adequate for a specific LAR item under evaluation or if additional precautions should be taken to address any relevant specific LAR item or dry storage system characteristics, as well as effects related to the combination of multiple individual changes.

Based on past LARs, and in coordination with NRC project management, a set of common, individual LAR items were identified and included for evaluation within this tool by developing the associated risk estimations and rationales. However, the included LAR items cannot cover all possible future changes, and situations may arise in which the user of the tool wants to review an item that was not yet evaluated in this report. In such cases, the tool provides guidance for a consistent 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 with 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.

The LAR for dry storage systems, submitted by the certificate of compliance (CoC) holder or licensee to NRC, can include a request for a single or 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. Further, this tool is designed to evaluate the effect of individual changes on overall safety. LAR 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 review depth is necessary to accommodate the total level of risk.

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 only consider the main safety functions of a dry storage system—confinement, radiological protection, and criticality safety. The risk estimation process does not focus on meeting regulatory thresholds or requirements, although it is also important to recognize that the tool is not meant to challenge any U.S. regulatory requirements or thresholds or to challenge the standard review plan for spent fuel dry storage. Rather, the INL developed the tool to support NRC on conducting LARs through providing a risk-based approach. For instance, the tool’s risk evaluation methodology credits the fact that SNF rod cladding failure does not directly lead to a release of radioactive material to the environment. Current regulations require cladding integrity, and NRC reviewers need to evaluate the potential of an LAR item to cause cladding failure.

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 these systems use metal casks as the outermost structures (i.e., the overpack), while others use a different material, such as concrete structures. 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. Additionally, package operations, such as loading and transfer, are sometimes specific to the needs and requirements of the associated nuclear power plant. Also, different transfer cask designs are used. However, despite the variety of existing component designs and procedures 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 consider design-specific individual components whenever a generalization appeared feasible and appropriate, 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 structure outlined in NUREG/CR-6407 (J. W. McConnell 1996), and the authors refer to this document's component definitions for clarification. Additionally, 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.

- **Overpack:** If the overpack (typically metal or concrete) of a dry storage system is not part of the confinement barrier itself, it functions as the physical protection of the confinement barrier from mechanical and thermal loads and provides radiological protection of the environment (i.e., shielding).

- **Canister/Inner Cask:** The canister (e.g., the welded multipurpose canister) or (bolted) inner cask of a dry storage system represents the confinement barrier of a dry storage system. This structure is enclosed by the overpack. The SNF is placed inside the canister or inner cask. The functional difference between the canister and the inner cask is that the canister can be removed from the overpack, but the inner cask is permanently connected to the overpack.

- **Transfer Cask:** The transfer cask is utilized in the loading and transfer phase of dry storage operations to move the welded canister or bare SNF from the spent fuel pool to the overpack. It provides the 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 holds the SNF assemblies in place and provides criticality control by preventing SNF assemblies, which are placed in the basket cell, from deforming or relocating excessively.

2.3 Tree Diagram

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.

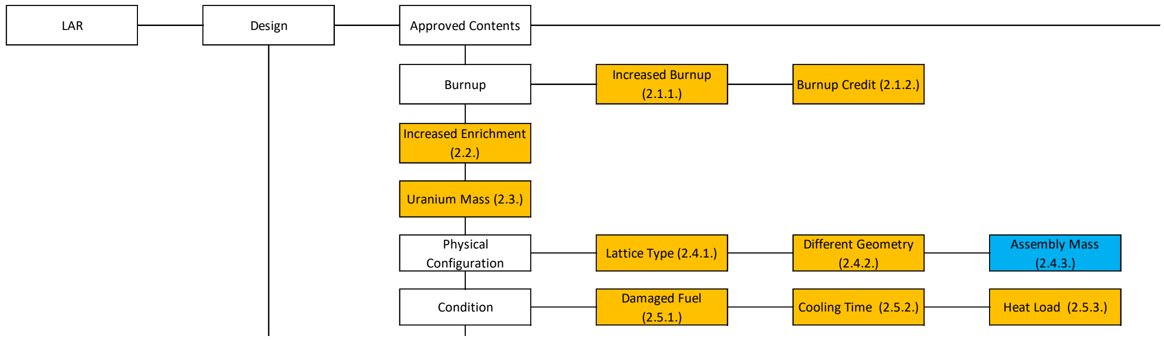

Figure 1. Example portion of Task 1 tree diagram.

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 argumentation 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 a change in the tree diagram is evaluated and categorized. The "LAR Review Recommendations" column 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, and is only applicable if a quantitative risk sensitivity study is possible.							

Table 2 provides a set of six rules that ensure the tool is properly used and 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 for a gate is possible, evaluate both gates independently.
3	Use the color scale to evaluate risk significance.
4	Review rationale document if gate color is blue. The gate number is provided in parenthesis.
5	Review rationale document for general background information or when uncertain about gate applicability.
6	Review rationale document if the gate is unevaluated.

2.4 Risk Significance Estimations

This report outlines the methodology used by the LAR risk significance determination tool, enabling the development of the accompanying tree diagram. The determination of the risk significance of the evaluated LAR items is based on semiquantitative (i.e., using available simulation or examination data and sensitivity studies on available PRA data) and qualitative assessments (i.e., via conducting subject matter expert 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).

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 the NRC or defined in coordination with NRC project management. The focus of the evaluations was on LARs pertaining to dry storage design modifications, regarding the chosen safety evaluation methods, concerning permit editorial modifications, or as to approve a different SNF content to be stored in dry storage systems. However, it is impossible to develop (and evaluate) a complete set of LAR items since potential future requests cannot be predicted. Thus, we have provided a strategy (see the Gate 9. Unevaluated Change section) to evaluate LAR items that are not included within this document.

It is important to note that this risk tool does not account for the effects of degradation, such as stress corrosion cracking. However, such effects could be included by expanding the tree diagram with a branch focused on LAR items related to dry storage system license extensions. The expandable structure of the tool allows for a permanent incorporation of such additional LAR items. Also, new fuel types, such as accident tolerant fuel, were not included due to the lack of currently available data.

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 lead to a release of radioactive material to the environment, to dry storage content criticality, or to a significant increase of the radiation exposure of the public or operating personnel. 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, 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. Thus, the assigned significances were determined considering the comprehensive set of criteria in an integrated fashion (Figure 2). The last column in Table 1 provides a basic review recommendation, associated with the risk significance estimation of an LAR item, to be developed by the NRC after extensive tool testing.

Figure 2. Integrated risk significance determination process.

Although the risk significant determination process could indicate the low risk significance of a specific LAR, the reviewer and applicant need to ensure the final design still meets all required regulatory requirements. The emphasis was put on the main dry storage functions, which include the protection of the public and personnel health. However, the process did not consider other functions, such as fuel retrievability or various functions related to system operations.

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 not as a static risk estimate. During the review and consideration of specific LAR item and dry storage system 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 3.

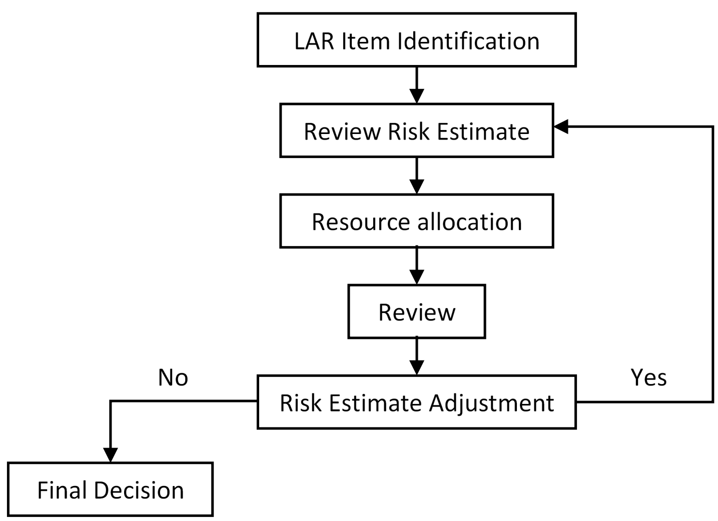

Figure 3. Review flowchart.

2.4.3 Available Quantitative Data and Literature

Figure 4 shows a summary of quantitative data used in an original PRA of a Holtec International HI-STORM 100 System, published in NUREG-1864 (Malliakos 2007). Although this PRA evaluates the risk of a dry storage operation with specific characteristics, and although its results are not directly applicable to other dry storage operations with different characteristics, it provides valuable insights into the risk profile of dry storage systems, and its data can be studied to obtain a general sense of risk related to a design change. Nevertheless, in the development of the present risk tool, these studies on the quantitative NUREG-1864 PRA data are only meant to support the risk significance estimation process of a specific LAR item, rather than determining it. For the determination, additional criteria, such as system redundancies, or safety margins, were considered using an integrated risk significance determination process (see Table 1 and Figure 2).

In the NUREG-1864 PRA, the quantitative data on risk is the probability of a latent cancer fatality for an individual within a 10-mile radius around the dry storage facility due to the release of radioactive isotopes and under consideration of a 20-year dry storage operation duration. The associated risk computed for such an event is 2.4E-12. Based on the probabilistic data of NUREG-1864, the risk sensitivity of a dry storage system to selected LAR items was studied, and the gained insight was used to support the development of selected rationales. However, instead of focusing on the cancer risk as a risk measure, it evaluated the increase of the probability of radioisotope release—such as fine SNF fragments, noble gases, or Chalk River Unidentified Deposit—from the dry storage system. This increase is computed by comparing the sum of the products of the accident probabilities (marked red in Table 3) and the probabilities of a release of radioactive isotopes (marked blue in Table 3) during each phase of the dry storage operation—as provided in NUREG-1864—with the sum of the products of the accident probabilities and the probabilities of a release of radioactive isotopes when considering the LAR. An example computation is presented in Table 4. The baseline canister release risk of the unmodified dry storage system is computed as 2.62E-04. This value can be compared to the canister release risk of a modified system, and such comparisons are used in some of the review risk significance rationales to get a better understanding of the sensitivity of the canister release risk to a specific LAR.

2.4.4 General Assumptions

For risk sensitivity evaluations that consider the probabilistic data provided by NUREG-1864 (Malliakos 2007), the following assumptions are made:

1.) The canister integrity of the dry storage system under consideration is comparable to the integrity of the multipurpose canister that is part of the Holtec International Storage Module (HI-STORM) 100 system analyzed in the NUREG-1864 PRA.

2.) The transfer cask integrity of the dry storage system under consideration is comparable to the integrity of the Holtec International Transfer Cask of the system analyzed in the NUREG-1864 PRA.

3.) The overpack integrity of the storage system under consideration is comparable to the integrity of the overpack of the HI-STORM 100 system analyzed in the NUREG-1864 PRA.

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

Table 3. HI-STORM 100 PRA data, adapted from NUREG-1864.

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
Handling	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-12	5.60E-17
Handling	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
Handling	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
Handling	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
Handling	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
Handling	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
Handling	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
Handling	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
Handling	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
Handling	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
Handling	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
Handling	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
Handling	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
Handling	18a	Lowering Transfer Cask onto Overpack (a)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	Noble Gas	1.00E+00	1.00E-10	1.12E-16
Handling	18b	Lowering Transfer Cask onto Overpack (b)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	Other than Noble Gas	1.50E-04	3.60E-04	6.05E-14
Handling	20a	Lifting Canister and Opening Transfer Lid (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	Noble Gas	1.00E+00	1.00E-10	1.57E-15
Handling	20b	Lifting Canister and Opening Transfer Lid (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	Other than Noble Gas	1.50E-04	3.60E-04	8.47E-13
Handling	21a	Transfering Canister to Overpack (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	Noble Gas	1.00E+00	1.00E-10	1.57E-15
Handling	21b	Transfering Canister to Overpack (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	Other than Noble Gas	1.50E-04	3.60E-04	8.47E-13
								Risk Handling Phase	**1.76E-12**
Transfer	26a	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (a)	Dropping Overpack	0.00E+00	0.00E+00	Noble Gas	0.00E+00	1.00E-10	0.00E+00
Transfer	26b	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (b)	Dropping Overpack	0.00E+00	0.00E+00	Other than Noble Gas	0.00E+00	3.60E-04	0.00E+00
Transfer	27a	Moving Storage Overpack to Preperation Area (a)	Dropping Overpack	0.00E+00	0.00E+00	Noble Gas	0.00E+00	1.00E-10	0.00E+00
Transfer	27b	Moving Storage Overpack to Preperation Area (b)	Dropping Overpack	0.00E+00	0.00E+00	Other than Noble Gas	0.00E+00	3.60E-04	0.00E+00
Transfer	28a	Holding Storage Overpack (a)	Dropping Overpack	0.00E+00	0.00E+00	Noble Gas	0.00E+00	1.00E-10	0.00E+00
Transfer	28b	Holding Storage Overpack (b)	Dropping Overpack	0.00E+00	0.00E+00	Other than Noble Gas	0.00E+00	3.60E-04	0.00E+00
Transfer	29a	Moving Storage Overpack away from Prepration Area (a)	Dropping Overpack	0.00E+00	0.00E+00	Noble Gas	0.00E+00	1.00E-10	0.00E+00
Transfer	29b	Moving Storage Overpack away from Prepration Area (b)	Dropping Overpack	0.00E+00	0.00E+00	Other than Noble Gas	0.00E+00	3.60E-04	0.00E+00
Transfer	30a	Transfering Storage Overpack on Asphalt to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	Noble Gas	0.00E+00	1.00E-10	0.00E+00
Transfer	30b	Transfering Storage Overpack on Asphalt to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	Other than Noble Gas	0.00E+00	3.60E-04	0.00E+00
Transfer	31a	Transfering Storage Overpack on Gravel to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	Noble Gas	0.00E+00	1.00E-10	0.00E+00
Transfer	31b	Transfering Storage Overpack on Gravel to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	Other than Noble Gas	0.00E+00	3.60E-04	0.00E+00
Transfer	32a	Moving Storage Overpack above Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	Noble Gas	0.00E+00	1.00E-10	0.00E+00
Transfer	32b	Moving Storage Overpack above Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	Other than Noble Gas	0.00E+00	3.60E-04	0.00E+00
Transfer	33a	Lowering Storage Overpack onto Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	Noble Gas	0.00E+00	1.00E-10	0.00E+00
Transfer	33b	Lowering Storage Overpack onto Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	Other than Noble Gas	0.00E+00	3.60E-04	0.00E+00
								Risk Transfer Phase	**0.00E+00**
Storage	34.1a	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	Noble Gas	1.00E+00	1.00E-10	7.00E-23
Storage	34.1b	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	Other than Noble Gas	1.00E+00	3.60E-04	2.52E-16
Storage	34.2a	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	Noble Gas	1.00E+00	1.00E-10	8.82E-21
Storage	34.2b	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	Other than Noble Gas	1.00E+00	3.60E-04	3.18E-14
Storage	34.3a	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	Noble Gas	1.00E+00	1.00E-10	3.50E-24
Storage	34.3b	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	Other than Noble Gas	1.00E+00	3.60E-04	1.26E-17
Storage	34.4a	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	Noble Gas	1.00E+00	1.00E-10	0.00E+00
Storage	34.4b	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	Other than Noble Gas	1.00E+00	3.60E-04	0.00E+00
								Risk Storage Phase (1-Year)	**3.20E-14**
								Risk 20 Years	**2.40E-12**

Table 4. HI-STORM 100 PRA, baseline release risk, adapted from NUREG-1864.

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
Handling	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
Handling	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	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
Handling	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
Handling	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	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
Handling	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	18a	Lowering Transfer Cask onto Overpack (a)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
Handling	18b	Lowering Transfer Cask onto Overpack (b)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
Handling	20a	Lifting Canister and Opening Transfer Lid (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	20b	Lifting Canister and Opening Transfer Lid (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21a	Transfering Canister to Overpack (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21b	Transfering Canister to Overpack (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase	**2.62E-04**
Transfer	26a	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	26b	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27a	Moving Storage Overpack to Preperation Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27b	Moving Storage Overpack to Preperation Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28a	Holding Storage Overpack (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28b	Holding Storage Overpack (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29a	Moving Storage Overpack away from Prepration Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29b	Moving Storage Overpack away from Prepration Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30a	Transfering Storage Overpack on Asphalt to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30b	Transfering Storage Overpack on Asphalt to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31a	Transfering Storage Overpack on Gravel to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31b	Transfering Storage Overpack on Gravel to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32a	Moving Storage Overpack above Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32b	Moving Storage Overpack above Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33a	Lowering Storage Overpack onto Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33b	Lowering Storage Overpack onto Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase	**0.00E+00**
Storage	34.1a	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
Storage	34.1b	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
Storage	34.2a	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.2b	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.3a	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.3b	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.4a	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
Storage	34.4b	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (1-Year)	**1.78E-10**
					Risk 20 Years	**2.62E-04**

Table 5. Categorization of SSCs according to safety, adapted from NUREG/CR-6407 (J. W. McConnell 1996).

Category	Criteria
A – Critical to safety	SSC failure could directly lead to loss of confinement, shielding, or criticality control.
B – Major importance to safety	SSC failure in conjunction with failure of another item could lead to loss of confinement, shielding, or criticality control.
C – Minor importance to safety	SSC failure 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 colors in this tree diagram provide information on the risk significance that was estimated for each considered change (i.e., LAR item) in the system. The gate numbers in the tree refer to sections in this document that provide rationales 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):

a. High (red-colored gates)

b. Medium (yellow-colored gates)

c. Low (green-colored gates)

The risk significance of some LAR items cannot be estimated without considering specific LAR item or system design details. The corresponding gates are marked in blue (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 review risk associated with 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 rationale of the review 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 and Inner Cask

2.5.1.1 Containment Shell

2.5.1.1.1 Gate 1.1.1.1.1. Shell Body

The shell body denotes the main part of the canister or inner cask. Its task is to uphold 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 (J. W. McConnell 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 for safety.

This conclusion can be evaluated based on available NUREG-1864 PRA data. The data indicates SNF handling as the critical operational phase, with respect to the risk of releasing radioactive isotopes. This phase contributes significantly to the overall risk during a 20-year operation, because the canister is not closed in this operational stage. The associated event initiation frequency is 5.60E-5, which is also the associate release risk under the assumption of a shell body failure due to a design flaw (see Table 7). However, canister modifications do not only affect the release risk during the SNF handling phase. Modifying the canister design could increase the risk of canister failure after closure under mechanical or thermal loading, too. Thus, the canister failure probability in accident scenarios other than SNF handling (i.e., dropping the canister-containing transfer cask) could increase, increasing the cumulative risk of a 20-year dry storage operation by a significant factor.

A reevaluation of the available PRA data that assumes a canister failure indicates an increase in risk for a dry storage operation of 20 years by a factor of almost 5 (see Table 7). This would point to a medium risk significance for an LAR item that affects an important component of the confinement barrier, when evaluated according to Criterion A in Table 1. However, no arguments for the large safety margins of the confinement barrier can be made. 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 high.

2.5.1.1.2 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 (J. W. McConnell 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 critical to safety. This conclusion can be evaluated by studying the effects on the release risk of modifying the confinement barrier design by using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

Table 7. HI-STORM 100 PRA, modified by 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
Handling	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
Handling	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	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
Handling	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
Handling	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
Handling	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	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
Handling	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
Handling	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	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
Handling	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	18a	Lowering Transfer Cask onto Overpack (a)	Dropping Transfer Cask, Height 100 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	18b	Lowering Transfer Cask onto Overpack (b)	Dropping Transfer Cask, Height 100 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	20a	Lifting Canister and Opening Transfer Lid (a)	Dropping Canister, Height 5.9 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	20b	Lifting Canister and Opening Transfer Lid (b)	Dropping Canister, Height 5.9 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	21a	Transfering Canister to Overpack (a)	Dropping Canister, Height 5.9 ft	5.60E-05	**1.00E+00**	5.60E-05
Handling	21b	Transfering Canister to Overpack (b)	Dropping Canister, Height 5.9 ft	5.60E-05	**1.00E+00**	5.60E-05
					Risk Handling Phase	**1.20E-03**
Transfer	26a	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	26b	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27a	Moving Storage Overpack to Preperation Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27b	Moving Storage Overpack to Preperation Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28a	Holding Storage Overpack (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28b	Holding Storage Overpack (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29a	Moving Storage Overpack away from Prepration Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29b	Moving Storage Overpack away from Prepration Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30a	Transfering Storage Overpack on Asphalt to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30b	Transfering Storage Overpack on Asphalt to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31a	Transfering Storage Overpack on Gravel to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31b	Transfering Storage Overpack on Gravel to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32a	Moving Storage Overpack above Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32b	Moving Storage Overpack above Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33a	Lowering Storage Overpack onto Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33b	Lowering Storage Overpack onto Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase	**0.00E+00**
Storage	34.1a	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	**1.00E+00**	7.00E-07
Storage	34.1b	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	**1.00E+00**	7.00E-07
Storage	34.2a	Storage	Overpack Struck by Aircraft	6.30E-09	**1.00E+00**	6.30E-09
Storage	34.2b	Storage	Overpack Struck by Aircraft	6.30E-09	**1.00E+00**	6.30E-09
Storage	34.3a	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.3b	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.4a	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	**1.00E+00**	3.70E-09
Storage	34.4b	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	**1.00E+00**	3.70E-09
					Risk Storage Phase (1-Year)	**1.42E-06**
					Risk 20 Years	1.23E-03
					Risk Increase Factor	**4.71E+00**

2.5.1.1.3 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 (J. W. McConnell 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 critical to safety. This conclusion can be evaluated by studying the effects of a modification of the confinement barrier design on the release risk using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

2.5.1.2 Openings

2.5.1.2.1 Gate 1.1.1.2.1. Lid Design

The lid 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 (J. W. McConnell 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 critical to safety. This conclusion can be evaluated by studying the effects of a modification of the confinement barrier design on the release risk using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

2.5.1.2.2 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 (J. W. McConnell 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 critical to safety. This conclusion can be evaluated by studying the effects of a modification of the confinement barrier design on the release risk using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

2.5.1.2.3 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 (J. W. McConnell 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 critical to safety. This conclusion can be evaluated by studying the effects of a modification of the confinement barrier design on the release risk

using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

2.5.1.3 Baseplate

2.5.1.3.1 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 (J. W. McConnell 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 critical to safety. This conclusion can be evaluated by studying the effects of a modification of the confinement barrier design on the release risk using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

2.5.1.4 Welds

2.5.1.4.1 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 therefore are rated as a Category A component according to NUREG/CR-6407 (J. W. McConnell 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, the welds are critical to safety. This conclusion can be evaluated by studying the effects of a modification of the confinement barrier design on the release risk using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

2.5.1.4.2 Gate 1.1.1.4.2. Welding Method

The welding methods used in the 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, the chosen welding method is critical to safety. This conclusion can be evaluated by studying the effects of a modification of the confinement barrier design on the release risk using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

2.5.1.5 Ports

2.5.1.5.1 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 (J. W. McConnell 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 critical to safety. This conclusion can be evaluated by studying the effects of a modification of the confinement barrier design on the release risk using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Thus, based on the discussion summarized above, any LAR item related to a design change of the confinement barrier requires the full attention of reviewer. The risk significance is high.

2.5.1.5.2 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 (J. W. McConnell 1996).

The confinement barrier and the leak check port plug of a canister are rated as Category A components according to NUREG/CR-6407 (J. W. McConnell 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 critical to safety. This conclusion is supported by evaluations of the effects of a modification of the confinement barrier design on the release risk using available NUREG-1864 PRA data. These evaluations indicate the potential for an increase of release risk by a factor of almost 5, as demonstrated in the rationale of Gate 1.1.1.1.1., which would indicate a medium risk. However, no arguments can be made for the large safety margins of the confinement barrier. Further, in case the port plug is required for adequate radiological protection, the safety margins are expected to be small. Thus, based on the discussion summarized above, any LAR item related to a design change of the leak check port requires the full attention of reviewer. The risk significance is high.

2.5.1.5.3 Gate 1.1.1.6. Backfill Pressure

The backfill pressure in a dry storage system is typically designed to reach pressures on 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 the necessary thermal heat transfer of the SNF and avoid an 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 could increase the internal pressure. A fire scenario could also result in over-pressurization. However, the pressure needs to reach levels high enough to cause a failure of the confinement barrier. 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.

2.5.1.6 Criticality Control

2.5.1.6.1 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 (J. W. McConnell 1996).

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, and likely, significantly reconfigured SNF. Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215 dry storage canisters showed that the content of most canisters remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as the SNF burnup) in the original, upper-bound criticality evaluations as published in the final safety analysis reports (FSARs) of dry storage systems. Further, 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). Furthermore, Alsaed (2018) suggests that, without significant moderation, commercial SNF cannot reach criticality. 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. Nevertheless, the analyses by 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to the ^{240}Pu decay.

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 safety margins are typically large. Criticality in a dry storage canister can only occur after exceeding significant criticality safety margins, and the concurrence of multiple adverse conditions, such as a significant reconfiguration of the fuel during a canister flooding event and significant degradation or dilution of neutron poisons in the canister structural components of water fill.

2.5.1.6.2 Gate 1.1.2.2. Neutron Poison Plates

Some dry storage canisters use neutron absorbers, such as neutron poison plates, to control the criticality of the content. These absorbers are classified as Category A components according to NUREG/CR-6407 (J. W. McConnell 1996).

The NRC's Division of Spent Fuel Storage and Transportation issued the Interim Staff Guidance-23 (ISG-23) regarding the technical considerations in crediting the neutron absorber content of metal matrix composites (US-NRC 2011) for criticality control in dry storage systems. This document refers to the American Standard for Testing and Materials (ASTM) Standard Practice C1671-07 as a technical guidance to the staff in evaluating the credit of neutron absorber content of neutron poison plates (ASTM C 1671-07 2007). ASTM noted that the criticality control function of neutron poison plates is only significant in the presence of a moderator, such as during the loading of fuel under water, or water ingress

resulting from hypothetical accident conditions. Further, the ASTM standard describes the qualification of neutron absorber materials in three aspects:

(1) durability for the intended service

(2) physical characteristics meeting the design requirements

(3) uniformity of ^{10}B distribution in the neutron absorber material.

ISG-23 added guidance to test the plates' physical integrity after submersion and drying. The concern underlying this additional test is not criticality control but rather the presence of blistering after canister drying, which may have an adverse impact on fuel retrievability.

The addition of neutron poison plates may increase the approved content of dry storage systems in terms of fuel enrichment. However, nonuniformity or a physical deterioration of the poison plates may increase the risk of unintended criticality during wet loading operations. In certain cases, the dry storage certification applicants may use lower-end ^{10}B concentrations in associated safety evaluations to account for the possible range of nonuniformity accompanied with technical justifications for selecting a lower-end value, such as only modeling 90% of the design ^{10}B concentration (TransNuclear, Inc. 1999). NRC staff may need to review the safety evaluations and models to minimize the risk of local hot spots (i.e., localized areas of high neutron densities) enabling criticality accidents.

Related to risk of unintended criticality (e.g., due to an evaluation error), it is important to point out that criticality in a dry storage cask requires a moderated system, and likely, significantly reconfigured SNF. Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215 dry storage canisters showed that the content of most canisters remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to take recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay.

The available literature and data lead to the qualitative determination that the risk significance of a modification of neutron poison plates with respect to an original design is medium. Criticality safety margins are typically large, but there remains a risk of criticality accidents due to the nonuniformity of ^{10}B in neutron poison plates. However, criticality in a dry storage canister can only occur after exceeding significant criticality safety margins and the concurrence of multiple adverse conditions, such as a significant reconfiguration of the fuel during a canister flooding event and significant degradation or dilution of neutron poisons in the canister structural components of water fill.

2.5.1.7 Shielding

2.5.1.7.1 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 the canister lid welding or when moving the canister into the overpack for storage, before closing the shielded lid or door of the storage overpack. The shield plug is classified as a Category A component according to NUREG/CR-6407 (J. W. McConnell 1996). There is no redundancy incorporated in a typical dry storage design, and thus, a modification of the shield plug could lead to a significantly increased dose rate received by the operating personnel. 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 high.

2.5.1.8 Structural Integrity

2.5.1.8.1 Gate 1.1.4.1. Canister Hardware

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

The function of the canister hardware is limited to structural integrity (J. W. McConnell 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.

2.5.1.9 Operations Support – Lifting

2.5.1.9.1.1 Gate 1.1.5.1.1. Canister Lugs, Trunnions, and 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 components according to NUREG/CR-6407 (J. W. McConnell 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, they play an important role during the canister transfer from the transfer cask into the storage overpack, since they have a controlling effect on the initiating accident frequency in the corresponding SNF handling phases, as shown in the NUREG-1864 PRA (Malliakos 2007).

Modifying the design of the lugs, trunnions, or grapples could increase the probability of a lifting accident to a value above 5.6E-5 (which is the probability of a lifting accident in the NUREG-1864 PRA, see Table 8). A reevaluation of the available PRA data indicates that, if a canister is accidentally dropped due to a failure of the lugs, trunnions, or grapples in the appropriate handling phases, it could increase the canister release risk for a dry storage operation of 20 years by over 4,000 times (see Table 8), compared to the original system condition. Based on this evaluation, the risk significance of an LAR item that includes a modification of the canister lugs, trunnions, or grapples is high.

Table 8. HI-STORM 100 PRA, modified by 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
Handling	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
Handling	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	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
Handling	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
Handling	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	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
Handling	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	18a	Lowering Transfer Cask onto Overpack (a)	Dropping Transfer Cask, Height 100 ft	5.60E-04	2.00E-02	1.12E-05
Handling	18b	Lowering Transfer Cask onto Overpack (b)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
Handling	20a	Lifting Canister and Opening Transfer Lid (a)	Dropping Canister, Height 5.9 ft	1.00E+00	2.80E-01	2.80E-01
Handling	20b	Lifting Canister and Opening Transfer Lid (b)	Dropping Canister, Height 5.9 ft	1.00E+00	2.80E-01	2.80E-01
Handling	21a	Transfering Canister to Overpack (a)	Dropping Canister, Height 5.9 ft	1.00E+00	2.80E-01	2.80E-01
Handling	21b	Transfering Canister to Overpack (b)	Dropping Canister, Height 5.9 ft	1.00E+00	2.80E-01	2.80E-01
					Risk Handling Phase	**1.12E+00**
Transfer	26a	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	26b	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27a	Moving Storage Overpack to Preperation Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27b	Moving Storage Overpack to Preperation Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28a	Holding Storage Overpack (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28b	Holding Storage Overpack (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29a	Moving Storage Overpack away from Prepration Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29b	Moving Storage Overpack away from Prepration Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30a	Transfering Storage Overpack on Asphalt to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30b	Transfering Storage Overpack on Asphalt to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31a	Transfering Storage Overpack on Gravel to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31b	Transfering Storage Overpack on Gravel to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32a	Moving Storage Overpack above Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32b	Moving Storage Overpack above Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33a	Lowering Storage Overpack onto Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33b	Lowering Storage Overpack onto Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase	**0.00E+00**
Storage	34.1a	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
Storage	34.1b	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
Storage	34.2a	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.2b	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.3a	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.3b	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.4a	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
Storage	34.4b	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (1-Year)	**1.78E-10**
					Risk 20 Years	**1.12E+00**
					Risk Increase Factor	**4.28E+03**

2.5.1.9.1.2 Gate 1.1.5.1.2. Canister Lug, Trunnion, and 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 themselves. These components are classified as Category A components according to NUREG/CR-6407 (J. W. McConnell 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, they play an important role during the canister transfer from the transfer cask into the storage overpack, since they have a controlling effect on the initiating accident frequency in the corresponding SNF handling phases as shown in the NUREG-1864 PRA (Malliakos 2007). Modifying the design of the welds or bolts of the lugs, trunnions, or grapples, could affect the functionality of the lugs, trunnions, or grapples, and thus, increase the canister release risk considering a dry storage operation of 20 years by over 4,000 times, as demonstrated in the rationale of Gate 1.1.5.1.1. Based on this evaluation, the risk significance of an LAR item that includes a modification of the welds or bolts of canister lugs, trunnions, or grapples is high.

2.5.1.10 Gate 1.1.6. Canister Type

A new canister type requires a thorough reevaluation of the dry storage system with respect to criticality control, confinement, shielding capability, structural integrity, thermal performance, and operations. An error in any of the evaluations could 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 high.

2.5.2 Design—Overpack

2.5.2.1 Shielding

2.5.2.1.1 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 incorporated in a metallic overpack. 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 (J. W. McConnell 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, 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 radiation exposure 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, there is a good chance that radiation exposure of 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.

Following NUREG-2215 (US-NRC 2020), Section 8.5.8.2, the NRC accepts the concrete temperature limits in American Concrete Institute (ACI) 349, "Code Requirements for Nuclear Safety-Related Concrete Structures." (ACI 2014) Although ACI 349 is a structural code, NRC generally accepts its use to provide a reasonable assurance of shielding performance as well.

Section E.4 of ACI 349 provides acceptable limits for normal, short-term, and accident conditions. NRC has received applications that propose using concrete overpacks exposed to temperatures above these ACI 349 limits. Such a proposal requires a medium degree of review, for the following reasons:

1. There is a lack of historical operating experience on elevated temperature scenarios.
2. For that reason, the applicant needs to provide data on the structural and shielding performance and the relevant analysis showing that the concrete overpack meets ACI 349 criteria after exposure to elevated temperatures.
3. Possible implications of concrete overpack usage in these scenarios include:
    a. A considerable reduction of strength, such as a 35% compressive strength reduction at 572°F for 2 days (M.K. Kassir 1996)
    b. A considerable deterioration of radiation shielding performance, such as measured significant increases in neutron flux through concrete as water was lost during elevated temperature exposure (Peterson 1960).
4. The unknown performance of concrete in these scenarios requires a significant level of review effort.

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

2.5.2.1.2 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 (J. W. McConnell 1996).

Dose limits to the public and 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 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 radiation exposure 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, there is a good chance that radiation exposure of 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.

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Section E.4 of ACI 349 provides acceptable limits for normal, short-term, and accident conditions. The NRC has received applications that propose the use of concrete overpacks exposed to temperatures above these ACI 349 limits. Such a proposal requires a medium degree of review, for the following reasons:

1. There is a lack of historical operating experience on the elevated temperature scenarios.
2. For that reason, the applicant needs to provide data on the structural and shielding performance and the relevant analysis showing that the concrete overpack meets ACI 349 criteria after exposure to elevated temperatures.
3. Possible implications of concrete overpacks usage in these scenarios include:
    a. A considerable reduction of strength, such as a 35% compressive strength reduction at 572°F for 2 days (M.K. Kassir 1996)
    b. A considerable deterioration of radiation shielding performance, such as measured significant increases in neutron flux through concrete as water was lost during elevated temperature exposure (Peterson 1960).
4. The unknown performance of concrete in these scenarios require a significant level of review effort.

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

2.5.2.1.3 Gate 1.2.1.3. Shielded Access Door

Some concrete dry storage systems use a shielded access door to close the concrete overpack after canister loading. 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 a Category B component (i.e., a failure would be a major safety concern) according to NUREG/CR-6407 (J. W. McConnell 1996).

Dose limits to the public and 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 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 radiation exposure 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, there is a good chance that radiation exposure of 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. The risk significance of an LAR item that includes a modification of the access door is medium.

2.5.2.2 Heat Transfer

2.5.2.2.1 Gate 1.2.2.1. Fins

Dry storage systems use components such as fins to transfer the 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 (J. W. McConnell 1996), which indicates that they play an important role ensuring the safety of a dry storage system.

However, thermal simulations of cask storage systems appear to provide very conservative data (Hanson, et al. 2016). For instance, temperature data recorded during backfilling gas experiments in a CASTOR V/21 cask loaded in 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 (Dziadosz, et al. 1986). Further, newer simulations and experimental measurements (Fort, et al. 2019, Csontos 2020) indicate that the peak cladding temperatures of high-burnup SNF rods likely do not exceed the permitted thresholds. Nevertheless, the applicant needs to ensure compliance of the peak cladding temperature with the regulatory thermal limits via conducting a thermal reassessment of the dry storage system. Note that the regulatory 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 that latter effects might not play as important a role as previously thought.

The effectiveness of fins for temperature control during a fire accident could be certainly contested, and some cask vendors consider a loss of neutron shielding during a fire in their safety evaluations already.

Finally, fuel rod failure alone, such as due to high temperatures, would not lead to criticality, a release of radioactive material to the environment, or raise the radiation exposure of the public or operation personnel significantly. However, redundancy due to the confinement barrier provided by the barrier cannot be claimed, since a flawed fin design affects both the integrity of the SNF rods and the canister. Furthermore, the discrepancy between simulations and empirically collected data suggest a complex evaluation process, with a significant probability of an error being undetected. Based on the discussion above, the risk significance of a change in fin design is medium.

2.5.2.3 Vents

2.5.2.3.1 Gate 1.2.2.2.1. Vent Openings

In NUREG-1864 (Malliakos 2007), the long-term blockage of vents in a HI-STORM 100 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 that about 50% of all fuel rods could experience peak cladding temperatures above the long-term temperature threshold of 400°C, 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 would not likely 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. Nevertheless, fuel rod failure alone

would not lead to criticality, a release of radioactive material to the environment, or raise the radiation exposure of the public or operation personnel to above acceptable levels, although redundancy cannot be claimed, since such a scenario would also challenge the confinement barrier.

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

2.5.2.3.2 Gate 1.2.2.2.2. Vent Channels

In NUREG-1864 (Malliakos 2007), the long-term blockage of vents in a HI-STORM 100 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 that about 50% of all fuel rods could experience peak cladding temperatures above the long-term temperature threshold of 400°C, 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. Nevertheless, fuel rod failure alone would not lead to a release of radioactive material to the environment, lead to criticality, or raise the radiation exposure of the public or operation personnel to above acceptable levels, although redundancy cannot be claimed, since such a scenario would challenge the confinement barrier as well.

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

2.5.2.4 Structural Integrity

2.5.2.4.1 Gate 1.2.3.1. Overpack

NUREG/CR-6407 (J. W. McConnell 1996) classified the overpack 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 overpack is the system design characteristic that determines whether or not it provides structural support to the gamma shield (J. W. McConnell 1996). In the latter case, a modification of the shell could lead to a quick loss of gamma shielding during a fire event.

Following NUREG-2215 (US-NRC 2020), Section 8.5.8.2, the NRC accepts the concrete temperature limits in ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures" (ACI 2014). Although ACI 349 is a structural code, the NRC generally accepts its use to provide a reasonable assurance of shielding performance as well.

Section E.4 of ACI 349 provides acceptable limits for normal, short-term, and accident conditions. The NRC has received applications that propose using concrete overpacks exposed to temperatures above these ACI 349 limits. Such a proposal requires a medium degree of review, for the following reasons:

1. There is a lack of historical operating experience on the elevated temperature scenarios.

2. For that reason, the applicant needs to provide data on the structural and shielding performance and the relevant analysis showing that the concrete overpack meets ACI 349 criteria after exposure to elevated temperatures.
3. Possible implications of concrete overpacks usage in these scenarios include:
   a. A considerable reduction of strength, such as a 35% compressive strength reduction at 572°F for 2 days (M.K. Kassir 1996)
   b. A considerable deterioration of radiation shielding performance, such as measured significant increases in neutron flux through concrete as water was lost during elevated temperature exposure (Peterson 1960)
4. The unknown performance of concrete in these scenarios require a significant level of review effort.

Some dry storage canisters are also licensed for transportation, and the overpack 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 overpack 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 (Malliakos 2007), a reduction of the protection provided by the overpack (e.g., due to a design change) could increase the probability of the release of radionuclides and noble gas during an earthquake or airplane impact. The maximum possible increase of the overall risk to the public for a 20-year dry storage operation is only 11% (see Table 9), due to the low initiating event frequencies during storage. However, system redundancy cannot be claimed, since it is expected that the confinement barrier is much weaker than the overpack and could fail as a consequence of overpack failure. Further, a modification of the overpack 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 overpack is medium.

2.5.2.4.2 Gate 1.2.3.2. Access Door and 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 (J. W. McConnell 1996), and the bolts are necessary for holding the shield in place.

Dose limits to the public and 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 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 radiation exposure 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, there is a good chance that radiation exposure of 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. The risk significance of an LAR item that includes a modification of the access door bolts is medium.

Table 9. HI-STORM 100 PRA, modified by 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
Handling	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
Handling	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	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
Handling	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
Handling	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	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
Handling	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	18a	Lowering Transfer Cask onto Overpack (a)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
Handling	18b	Lowering Transfer Cask onto Overpack (b)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
Handling	20a	Lifting Canister and Opening Transfer Lid (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	20b	Lifting Canister and Opening Transfer Lid (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21a	Transfering Canister to Overpack (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21b	Transfering Canister to Overpack (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase	**2.62E-04**
Transfer	26a	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	26b	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27a	Moving Storage Overpack to Preperation Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27b	Moving Storage Overpack to Preperation Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28a	Holding Storage Overpack (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28b	Holding Storage Overpack (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29a	Moving Storage Overpack away from Prepration Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29b	Moving Storage Overpack away from Prepration Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30a	Transfering Storage Overpack on Asphalt to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30b	Transfering Storage Overpack on Asphalt to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31a	Transfering Storage Overpack on Gravel to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31b	Transfering Storage Overpack on Gravel to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32a	Moving Storage Overpack above Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32b	Moving Storage Overpack above Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33a	Lowering Storage Overpack onto Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33b	Lowering Storage Overpack onto Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase	**0.00E+00**
Storage	34.1a	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	**1.00E+00**	7.00E-07
Storage	34.1b	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	**1.00E+00**	7.00E-07
Storage	34.2a	Storage	Overpack Struck by Aircraft	6.30E-09	**1.00E+00**	6.30E-09
Storage	34.2b	Storage	Overpack Struck by Aircraft	6.30E-09	**1.00E+00**	6.30E-09
Storage	34.3a	Storage	Overpack Struck by Meteorite	3.50E-14	**1.00E+00**	3.50E-14
Storage	34.3b	Storage	Overpack Struck by Meteorite	3.50E-14	**1.00E+00**	3.50E-14
Storage	34.4a	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	**1.00E+00**	3.70E-09
Storage	34.4b	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	**1.00E+00**	3.70E-09
					Risk Storage Phase (1-Year)	**1.42E-06**
					Risk 20 Years	2.90E-04
					Risk Increase Factor	**1.11E+00**

2.5.2.4.3 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 in NUREG/CR-6407 (J. W. McConnell 1996). 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 during such an event if not properly designed.

In the NUREG-1864 PRA (Malliakos 2007), the earthquake acceleration that could lead to a cask tipping over is an earthquake causing a ground acceleration of 1.35 g. The corresponding annual event frequency is 7E-7. Although hypothetical, a poorly designed storage pad could reduce the ground acceleration necessary to tip over the cask, and a more frequent earthquake with a lower ground acceleration could be sufficiently strong enough to lead to such an event. Such an earthquake could have 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 assuming an event frequency of a cask tipping over due to an earthquake increased 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 item, and the risk significance for an LAR item that includes the modification of the storage pad is low.

2.5.2.4.4 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 (J. W. McConnell 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 (Malliakos 2007) 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 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 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 for example a concrete 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 in a collapse scenario can be expected to be softer compared to a canister impact caused by a drop during transfer, due to the anticipation of a lower drop height and the anticipation that some of the kinetic energy is going to be dissipated by 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 medium.

2.5.2.4.5 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 (J. W. McConnell 1996). It is part of the structure of a dry storage system but 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 low.

Table 10. HI-STORM 100 PRA, modified by 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
Handling	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
Handling	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	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
Handling	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
Handling	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	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
Handling	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	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
Handling	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E-06	5.60E-11
Handling	18a	Lowering Transfer Cask onto Overpack (a)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
Handling	18b	Lowering Transfer Cask onto Overpack (b)	Dropping Transfer Cask, Height 100 ft	5.60E-05	2.00E-02	1.12E-06
Handling	20a	Lifting Canister and Opening Transfer Lid (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	20b	Lifting Canister and Opening Transfer Lid (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21a	Transfering Canister to Overpack (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21b	Transfering Canister to Overpack (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase	**2.62E-04**
Transfer	26a	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	26b	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27a	Moving Storage Overpack to Preperation Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27b	Moving Storage Overpack to Preperation Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28a	Holding Storage Overpack (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28b	Holding Storage Overpack (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29a	Moving Storage Overpack away from Prepration Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29b	Moving Storage Overpack away from Prepration Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30a	Transfering Storage Overpack on Asphalt to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30b	Transfering Storage Overpack on Asphalt to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31a	Transfering Storage Overpack on Gravel to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31b	Transfering Storage Overpack on Gravel to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32a	Moving Storage Overpack above Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32b	Moving Storage Overpack above Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33a	Lowering Storage Overpack onto Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33b	Lowering Storage Overpack onto Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase	**0.00E+00**
Storage	34.1a	Storage	Overpack Tip-over due to Seismic Event	**7.00E-04**	1.00E-06	7.00E-10
Storage	34.1b	Storage	Overpack Tip-over due to Seismic Event	**7.00E-04**	1.00E-06	7.00E-10
Storage	34.2a	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.2b	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.3a	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.3b	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.4a	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
Storage	34.4b	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (1-Year)	**1.58E-09**
					Risk 20 Years	2.62E-04
					Risk Increase Factor	**1.00E+00**

2.5.2.5 Operations Support

2.5.2.5.1 Gate 1.2.4.1. Access Door Lifting Lugs

Unlike the shielded access door or lid, the lifting lugs are operations support items. They are needed for a successful opening or closing of the door or lid. A failure could lead to a delay in cask closure, and consequently, to an increase in the radiation exposure of the operating personnel. However, the radiation exposure of loading and operating personnel is actively monitored and typically recorded with personal radiation dosimeters. There is a good chance that radiation exposure of 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 the modification of the access door lifting lugs is medium, which aligns with the Category B rating of NUREG/CR-6407 (J. W. McConnell 1996).

2.5.2.5.2 Gate 1.2.4.2. Protective Cover

Some casks designs (like the Transnuclear TN-32) include a protective cover to further protect the cask from snow, ice, and dust. This item is considered a Category C item according NUREG/CR-6407 (J. W. McConnell 1996). However, the FSAR of the TN-32 classifies the protective cover as an SSC important to safety, and the analyses in the report consider 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. Nevertheless, 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 medium.

2.5.2.6 Lifting

2.5.2.6.1 Gate 1.2.4.3.1. Overpack Lugs, Trunnions, and Grapples

Overpack lugs, trunnions, or grapples refer to the most outer attachments used to lift and move the dry storage cask. The risk significance of the LAR item that includes a modification of these components is directly related to whether the overpack is moved after it was loaded (e.g., within a secondary containment) with SNF or if the equipment is only used for moving the overpack during installation at the storage pad for long-term storage. Further, the large safety margins (e.g., an overpack has a good chance of surviving severe accidents, such as an airplane impact) need to be considered. If the overpack is moved after loading, the risk associated with a modification of the lugs, trunnions, or grapples is medium, since it could lead to an increase in the drop frequency of the cask and consequently to an increased probability of a release of radioactive material to the environment. However, if the overpack is only moved in an empty state, the risk significance of an LAR item that includes a modification of the overpack lugs, trunnions, or grapples is low.

2.5.2.6.2 Gate 1.2.4.3.2. Overpack Lugs, Trunnions, Grapples Bolts or Welds

The safe lifting of the overpack is directly affected by the strength of the connection (e.g., bolts or welds) between the lifting lugs, trunnions, or grapples and the overpack. It is important that the load-bearing joint is not necessarily located between the overpack and the attachment but 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 only lifted in an empty state or also after loading.

Further, the large safety margins (e.g., an overpack has a good chance of surviving severe accidents, such as an airplane impact) need to be considered. If the overpack is moved after loading, the risk associated with a modification of the lug, trunnion, or grapple bolts or welds is 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. However, if the overpack is only moved while empty, the risk significance of an LAR item that includes a modification of the overpack lug, trunnion, or grapple welds or bolts is low.

2.5.2.6.3 Gate 1.2.4.4. Security Lockwire and Seals

Security lockwires and seals are classified as Category C items in NUREG/CR-6407 (J. W. McConnell 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 low.

2.5.2.6.4 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 protect more sensitive parts of the overpack from weather exposure. However, this shell does not fulfill any safety function and is classified as Category C items in NUREG/CR-6407 (J. W. McConnell 1996). Thus, the risk associated with a modification of the shielding shell is low.

2.5.2.7 Other

2.5.2.7.1 Gate 1.2.5.1. Lightning Protection

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

In NUREG-1864 (Malliakos 2007), 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 Oak Ridge National Laboratory researchers 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 nearby workers being struck by lightning.

2.5.3 Design—Transfer Cask

2.5.3.1 Shielding

2.5.3.1.1 Gate 1.3.1.1. Water Jacket

The transfer cask, including its components, is classified as a Category B item in NUREG/CR-6407 (J. W. McConnell 1996).

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

personal dosimeters. The 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, immediately exposing personnel to high radiation.

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

2.5.3.1.2 Gate 1.3.1.2. Lead Shield

The transfer cask, including its components, is classified as a Category B item in NUREG/CR-6407 (J. W. McConnell 1996).

Transfer casks typically have a lead shield that provides protection from gamma radiation. A modification of this shield could increase the 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 provides physical protection to the canister. Therefore, any modification of the shell of the transfer cask, including the lead shield, could increase the risk of a confinement failure if the loaded transfer cask is dropped during handling.

A conservative evaluation of the risk increase due to such a change includes an assumption of canister failure (i.e., a probability of release of radioactive nuclides) during canister transfer. A reevaluation of the NUREG-1864 PRA under consideration of this assumption indicates an increase of the total release risk by a factor of 4 (see Table 11).

Based on the discussion summarized above, the risk significance of an LAR item that includes a modification of the gamma shield is medium.

2.5.3.1.3 Gate 1.3.1.3. Top Lid

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

Transfer casks typically have a top lid that provides protection from radiation. A modification of this lid could increase the 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. Nevertheless, the transfer cask provides physical protection to the canister. Therefore, any modification of the shell of the transfer cask, including the top lid, could increase the risk of a confinement failure if the loaded transfer cask is dropped during handling.

Therefore, any modification of the shell of the transfer cask, including the top lid, could increase the risk of a confinement failure if 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 larger than 5 (see 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 top lid is medium.

Table 11. HI-STORM 100 PRA, modified by Gate 1.3.1.1.

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
Handling	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	5.60E-05	1.00E+00	5.60E-05
Handling	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
Handling	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
Handling	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
Handling	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
Handling	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
Handling	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
Handling	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
Handling	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
Handling	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
Handling	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
Handling	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
Handling	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	5.60E-05	1.00E+00	5.60E-05
Handling	18a	Lowering Transfer Cask onto Overpack (a)	Dropping Transfer Cask, Height 100 ft	5.60E-05	1.00E+00	5.60E-05
Handling	18b	Lowering Transfer Cask onto Overpack (b)	Dropping Transfer Cask, Height 100 ft	5.60E-05	1.00E+00	5.60E-05
Handling	20a	Lifting Canister and Opening Transfer Lid (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	20b	Lifting Canister and Opening Transfer Lid (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21a	Transfering Canister to Overpack (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21b	Transfering Canister to Overpack (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase	**1.04E-03**
Transfer	26a	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	26b	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27a	Moving Storage Overpack to Preperation Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27b	Moving Storage Overpack to Preperation Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28a	Holding Storage Overpack (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28b	Holding Storage Overpack (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29a	Moving Storage Overpack away from Prepration Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29b	Moving Storage Overpack away from Prepration Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30a	Transfering Storage Overpack on Asphalt to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30b	Transfering Storage Overpack on Asphalt to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31a	Transfering Storage Overpack on Gravel to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31b	Transfering Storage Overpack on Gravel to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32a	Moving Storage Overpack above Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32b	Moving Storage Overpack above Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33a	Lowering Storage Overpack onto Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33b	Lowering Storage Overpack onto Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase	**0.00E+00**
Storage	34.1a	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
Storage	34.1b	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
Storage	34.2a	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.2b	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.3a	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.3b	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.4a	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
Storage	34.4b	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (1-Year)	**1.78E-10**
					Risk 20 Years	1.04E-03
					Risk Increase Factor	**3.99E+00**

2.5.3.1.4 Gate 1.3.1.4. Bottom Lid

The risk significance of a modification of the design of the bottom lid of the transfer cask (i.e., the pool lid or the canister-to-overpack transfer cask bottom lid) depends on whether or not the SNF handling operations or transfer phase includes 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 the drop and subsequent impact of an unprotected canister if a malfunction of the bottom lid occurred. Modifying the design of the transfer cask bottom lid could increase the probability of a lifting accident, which could, when combined with the possibility of a consequential canister failure, increase the canister release risk during a 20-year dry storage operation by a factor larger than 4,000 (see rationale of Gate 1.1.5.1.1.). Note that, generally, a drop of the unprotected canister due to a malfunction of the bottom lid is extraordinary unlikely.

If the canister does not rest on the bottom lid during any phase of the operations, the lid provides mainly radiological protection and structural protection during a transfer cask drop. Therefore, any modification of the shell of the transfer cask, including the bottom lid, could increase the risk of a confinement failure if 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 4 (see 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 is dependent on the cask operational procedures. It is 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 the event of a lid structural failure. If a canister drops as a consequence of bottom lid failure can be excluded, the risk significance is medium.

2.5.3.2 Shielding

2.5.3.2.1 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 (J. W. McConnell 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 if 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 4 (see 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 medium.

2.5.3.2.2 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 of Malliakos et al. (2007) indicates a potential risk increase by a factor of almost 4,000 (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 high.

Table 12. HI-STORM 100 PRA, modified by 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
Handling	3	Lifting Transfer Cask out of Cask Pit	Transfer Cask Drop into Cask Pit	1.00E+00	1.00E+00	1.00E+00
Handling	4	Lifting Transfer Cask over Railing	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
Handling	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
Handling	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
Handling	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
Handling	8	Lowering Transfer Cask onto Preparation Area	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
Handling	11	Lifting the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
Handling	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
Handling	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
Handling	14	Replacing Pool Lid by Transfer Lid	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
Handling	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
Handling	16	Holding the Transfer Cask	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
Handling	17	Moving Transfer Cask to Equipment Hatch	Dropping Transfer Cask, Height less than 1 ft	1.00E+00	1.00E-06	1.00E-06
Handling	18a	Lowering Transfer Cask onto Overpack (a)	Dropping Transfer Cask, Height 100 ft	1.00E+00	2.00E-02	2.00E-02
Handling	18b	Lowering Transfer Cask onto Overpack (b)	Dropping Transfer Cask, Height 100 ft	1.00E+00	2.00E-02	2.00E-02
Handling	20a	Lifting Canister and Opening Transfer Lid (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	20b	Lifting Canister and Opening Transfer Lid (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21a	Transfering Canister to Overpack (a)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
Handling	21b	Transfering Canister to Overpack (b)	Dropping Canister, Height 5.9 ft	5.60E-05	2.80E-01	1.57E-05
					Risk Handling Phase	**1.04E+00**
Transfer	26a	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	26b	Lifting Storage Overpack from Helman Rollers with Overpack Transporter (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27a	Moving Storage Overpack to Preperation Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	27b	Moving Storage Overpack to Preperation Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28a	Holding Storage Overpack (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	28b	Holding Storage Overpack (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29a	Moving Storage Overpack away from Prepration Area (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	29b	Moving Storage Overpack away from Prepration Area (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30a	Transfering Storage Overpack on Asphalt to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	30b	Transfering Storage Overpack on Asphalt to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31a	Transfering Storage Overpack on Gravel to Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	31b	Transfering Storage Overpack on Gravel to Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32a	Moving Storage Overpack above Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	32b	Moving Storage Overpack above Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33a	Lowering Storage Overpack onto Storage Pad (a)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
Transfer	33b	Lowering Storage Overpack onto Storage Pad (b)	Dropping Overpack	0.00E+00	0.00E+00	0.00E+00
					Risk Transfer Phase	**0.00E+00**
Storage	34.1a	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
Storage	34.1b	Storage	Overpack Tip-over due to Seismic Event	7.00E-07	1.00E-06	7.00E-13
Storage	34.2a	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.2b	Storage	Overpack Struck by Aircraft	6.30E-09	1.40E-02	8.82E-11
Storage	34.3a	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.3b	Storage	Overpack Struck by Meteorite	3.50E-14	1.00E+00	3.50E-14
Storage	34.4a	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
Storage	34.4b	Storage	Overpack Headed by Aircraft Fuel	3.70E-09	0.00E+00	0.00E+00
					Risk Storage Phase (1-Year)	**1.78E-10**
					Risk 20 Years	1.04E+00
					Risk Increase Factor	**3.97E+03**

2.5.3.2.3 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 (J. W. McConnell 1996).

The function of the hardware is limited to structural integrity (J. W. McConnell 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

2.5.4.1 Burnup

2.5.4.1.1 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 2012, Bidinger, et al. 2002). Thus, increasing the permitted burnup of the assemblies in a SNF dry storage system should reduce the reactivity of the canister 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, and likely, significantly reconfigured SNF. Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215 dry storage canisters showed that the content of most canisters remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that, without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay.

Dose limits to the public and 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 radiation exposure 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, there is a good chance

radiation exposure of 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.

The heat load of the SNF assemblies typically increases with higher burnup. If exceeding regulatory thresholds, it could lead to SNF degradation effects due to an increase in the peak cladding temperatures of the fuel rods, which could negatively affect the fuel rod conditions. However, the risk related to exceeding these limits is likely manageable.

Although it is important to recognize that 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 medium.

2.5.4.1.2 Gate 2.1.2. Burnup Credit

Criticality safety margins of current systems are typically large, but they could be reduced due to crediting the burnup in the criticality safety evaluations.

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, and likely, significantly reconfigured SNF. Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215 dry storage canisters showed that the content of most canisters remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay.

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

2.5.4.2 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 the 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, and likely, significantly reconfigured

SNF. Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215 dry storage canisters showed that the content of most canisters remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that, without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay.

Dose limits to the public and 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 radiation exposure 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, there is a good chance that radiation exposure of 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.

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

2.5.4.3 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, and likely, significantly reconfigured SNF. 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 the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to take recognize that the

canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay.

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 radiation exposure 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, there is a good chance that radiation exposure of 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. Based on the discussion above, the risk significance of an LAR item that includes modification of the permitted uranium mass per SNF assembly is medium.

2.5.4.4 Fuel Configuration

2.5.4.4.1 Gate 2.4.1. Lattice Type

The use of a different lattice type of SNF assemblies modifies the physical 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, and likely, significantly reconfigured SNF. 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 the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that, without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to take recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay. Based on the discussion above, the risk significance of an LAR item that includes a modification of the permitted SNF lattice type is medium.

2.5.4.4.2 Gate 2.4.2. Different Geometry

The inclusion of a different SNF assembly geometry modifies the physical 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, and likely, significantly reconfigured SNF. Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215 dry storage canisters showed that the content of most canisters remains subcritical by large safety margins

(K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that, without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to take recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay. Based on the discussion above, the risk significance of an LAR item that includes a modification of the permitted SNF assembly geometry is medium.

2.5.4.4.3 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 the overpack, including supporting structure. Potential consequences could be a loss of structural integrity of the fuel assemblies, a displacement of neutron poisons, or a basket failure, which could increase the reactivity in the cask if 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 low for small changes in the permitted assembly mass and medium-to-high for significant changes in the assembly mass.

2.5.4.5 Condition

2.5.4.5.1 Gate 2.5.1. Damaged Fuel

The current definition of damaged SNF is performance based (i.e., damaged SNF includes fuel rods or fuel assemblies that 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. Note that 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 an SNF cask to confine gross fuel particles, debris, or damaged assemblies to a known volume within a cask, to comply with criticality, thermal, shielding, and structural requirements, and to permit handling and retrievability (U.S. NRC 2007). Nevertheless, even if larger parts of the SNF breach the confinement of the can, the effects on safety of the dry storage system are likely insignificant. However, an accumulation of fissile material at the bottom of the canister or inner cask could affect 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, and likely, significantly reconfigured SNF. Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215

dry storage canisters showed that the content of most canisters remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that, without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to take recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay.

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 radiation exposure 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, there is a high chance that radiation exposure of 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.

Although the fuel cladding is the inner-most confinement barrier in a dry storage cask, the confinement capability of the canister or inner cask is adequate to prevent a release of radioactive material or noble gases to the environment. However, additional care needs to be taken for special cases, such as non-leak-tight systems (e.g., the Three-Mile Island [TMI] casks at INL).

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 is medium.

2.5.4.5.2 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 (e.g., the TMI casks at INL).

Dose limits to the public and 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 radiation exposure 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 there is a good chance that radiation

exposure of 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.

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.

2.5.4.5.3 Gate 2.5.3. Heat Load

The heat load in an 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, 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 also would challenge the confinement barrier. Additional care needs to be taken for special cases such as non-leak-tight systems (e.g., the TMI casks at INL).

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.

2.5.4.6 Gate 2.6. 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.

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, but the effect of a modification 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.). Further, cladding failure would not cause an immediate failure of any of the main safety functions of the dry storage system. For instance, for a release of radioactive material to the environment, 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. Note that additional care needs to be taken for special cases, such as non-leak-tight systems (e.g., the TMI casks at INL).

Related to the 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, it is important to note that criticality in a dry storage cask requires a moderated system, and likely, significantly reconfigured SNF. Although no quantitative PRA data on cask subcriticality is available, recent criticality evaluations of 215 dry storage canisters showed that the content of most canisters remains subcritical by large safety margins (K. Banerjee 2015). These margins can be traced back to typically uncredited, criticality-reducing effects (such as the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that, without significant moderation, commercial SNF cannot reach

criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay.

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 radiation exposure 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, there is a good chance that radiation exposure of 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.

An LAR item that includes a modification of the cask loading pattern requires a thermal reevaluation of the storage cask, and potentially, a reevaluation of criticality safety and the shielding capabilities of the system. The risk significance of such a modification depends on the potential increase in cladding temperature, and whether criticality-reducing effects of the original cask loading pattern were considered in the safety evaluations. The risk significance of such an item could range from low to medium.

2.5.4.7 Gate 2.7. 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.

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 provides sufficient confinement of the radionuclides and noble gases released from the rods. This assumption is supported by a recent phenomena identification and ranking table exercise for spent fuel cladding gross rupture completed by the Electric Power Research Institute (EPRI) (EPRI 2021). The study was jointly supported by the DOE, NRC and EPRI to define the regulatory term “gross rupture” identified in 10 CFR 72.122(h)(1) and to provide a practical guidance on performing fuel selection to meet the intent of NRC’s ISG-1 Rev. 2 (U.S. NRC 2007). The report details technical evaluation by a panel of experts on cladding gross rupture in spent fuel storage systems and transportation packages by considering thermal, radiological and criticality performance in hypothetical six scenarios. The panel defined the cladding gross rupture as a fuel cladding failure resulting in a fuel material release (of fragments of various sizes). Prevention of such failure would require precautions beyond those that are routinely used at sites during handling (i.e., fuel loading or unloading) of intact or canned fuel to prevent significant fuel material release. The six scenarios considered in the study include short-term fuel loading activities (i.e., loading, drying, and transfer), dry storage up to 60 years, long-term dry storage between 60 and 100 years, off-

normal conditions, unloading activities in wet or dry environments, and normal conditions of transport. The considered consequences included fission gas release from the fuel, fuel rod backfill gas release (i.e., helium) due to cladding breach, and fuel reconfiguration (i.e., release of coarse and aerosols from the fuel rod). The panel concluded that, under consideration of the three listed phenomena, all six scenarios were found to have a low significance on the thermal, radiological, and criticality safety performance of dry storage systems. More specifical, fuel reconfiguration—investigated for its potential to affect criticality safety—was also found to have a low significance on performance. The only scenario ranked as having a medium significance was unloading due to its possible influence on the radiological performance of the spent fuel package. However, EPRI's report assumes that the cask maintains confinement and containment and that evaluations considering accident scenarios during transportation were not included in this exercise. An expanded safety analysis is required to investigate the impacts of loss of confinement combined with spent fuel cladding gross rupture.

Related to the 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, and likely, significantly reconfigured SNF. 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 the SNF burnup) in the original, upper-bound criticality evaluations as published in the FSARs of dry storage systems. Further, the 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). Furthermore, Alsaed (2018) suggests that, without significant moderation, commercial SNF cannot reach criticality. 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. 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, and that crediting the SNF burnup in the criticality evaluations reduces the overall safety margin significantly. Additionally, it is important to take recognize that the canister is intentionally filled with water when placed into the spent fuel pool during SNF loading. Special care is required to ensure criticality safety during very long term (more than 100 years) dry storage, since the reactivity could increase due to ^{240}Pu decay.

It is important to recognize that the cladding is an important component in the multibarrier 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). But given the low probability of SNF rod failure due to a modification of the cladding material from one approved alloy to another approved alloy and the low chance of criticality, the risk significance of such a change is medium.

2.5.4.8 Gate 2.8. 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 public exposure 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 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, there is a good chance that radiation exposure of personnel and the public would be detected within a time frame that allows for putting corrective measures into place. Thus, the risk significance of an LAR item that includes the storage of SNF assemblies from another reactor type is medium.

2.5.4.9 Gate 2.9. 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 the reevaluation of the system by a reviewer. Nevertheless, insufficient shielding is likely going to be detected before significant consequences occur. For instance, the public exposure 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 radiation exposure 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, there is a good chance that radiation exposure of 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. 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 medium.

2.5.5 Evaluation

2.5.5.1 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 high.

2.5.5.2 Gate 3.2. Code and 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 medium.

2.5.6 Editorial Changes

2.5.6.1 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 low since such a change likely does not directly affect current or future dry storage operating procedures.

2.5.6.2 Revised Definitions and Clarifications

2.5.6.2.1 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.

2.5.6.2.2 Gate 4.2.2. Technical Specifications

The TSs are used by the operating personnel to define a cask loading pattern or operating procedures. Consequently, any change in the TSs requires a careful review.

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

2.5.6.2.3 Gate 4.2.3. CoC

The risk significance of an LAR item that includes the revision of definitions and provision of clarifications in the CoC is high since such a change could directly affect current dry storage operating procedures. The impact of the change must be evaluated because of the significance of the CoC document.

2.5.7 Unevaluated Change

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 developed tree diagram should provide the user with the information required to assess the risk associated with the review of a 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 and 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 a 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 upon user experience from pilot studies will help identify additional opportunities for improvement, especially on the workflow when using this tool. Additionally, 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 would improve the fidelity of the existing risk tool. The construction of a new SNF dry storage system PRA and an SNF transportation system PRA 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. PRAs could be designed for all portions of the used fuel cycle, including wet storage, cask loading, and onsite dry cask storage, as well as the transportation of the dry casks, to fully assess the areas of elevated risk to the general public.

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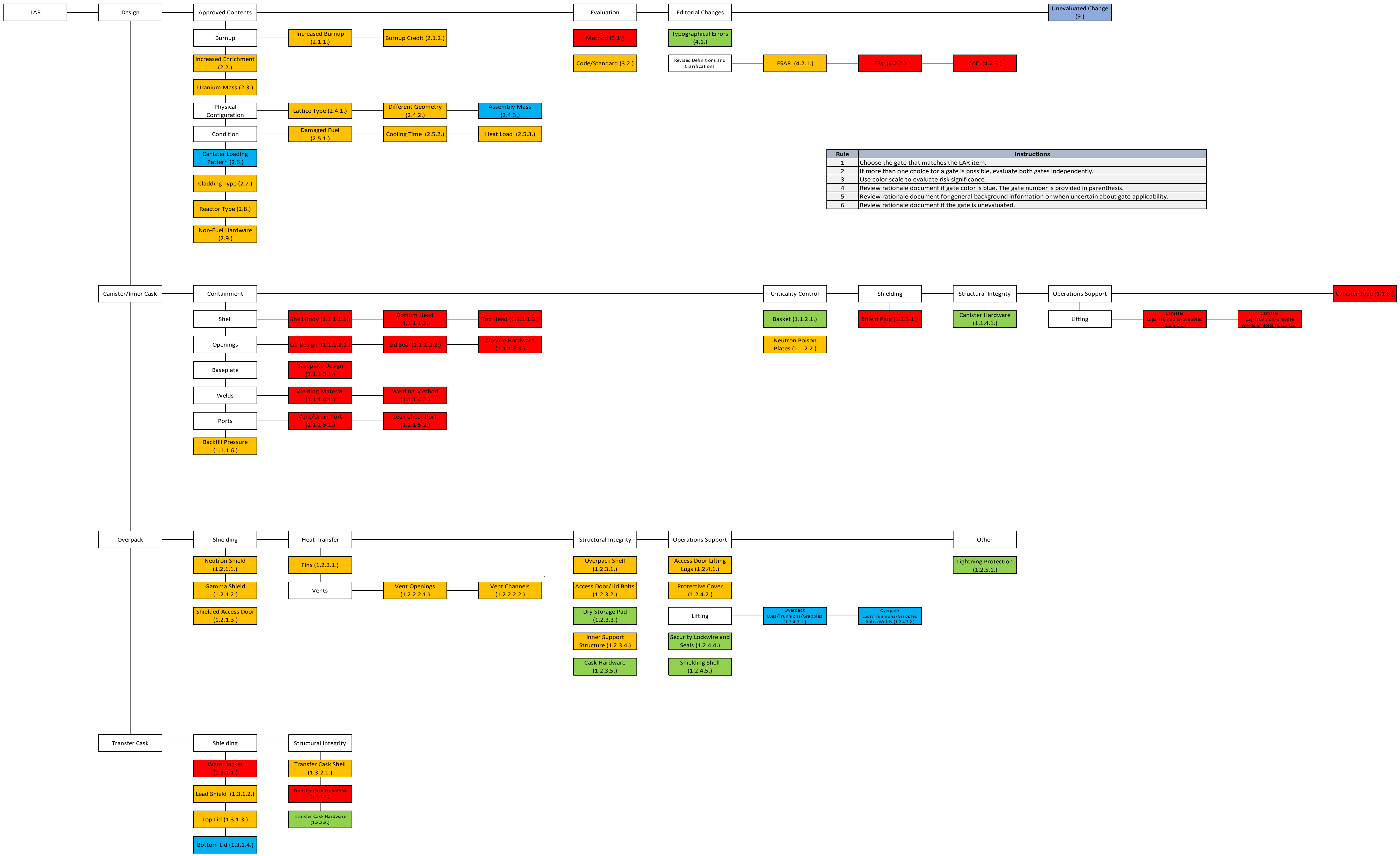
APPENDIX

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, and 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. The gate number is provided in parenthesis.
5	Review rationale document for general background information or when uncertain about gate applicability.
6	Review rationale document if the gate is unevaluated.