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Design Calculation or Analysis Cover Sheet

Complete only applicable items.

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Intra-Site Operations and BOP Reliability and Event Sequence Categorization Analysis

6. Group
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8. Notes/Comments

This revision addresses the following condition reports (CRs): 11966, 11989, 12002, 12101, 12103, 12105, 12176, 12245, and 12310. Substantive changes affect the sections listed below. In addition, editorial changes (including updating headers and footers) have been made throughout and affect every page.

- Main body sections: 1, 6.2.2.3.6, 6.3.1.3, 6.3.1.5, 6.3.2.2, 6.7.2, 6.9.1.
- Attachment A: [A-] 4.1.2.
- Attachment B: [B-] 1.5.1.5, 1.5.1.7, 1.5.2.5, 2.2.1, 2.4.1.5, 2.4.2.2, 2.4.2.3, 2.4.2.5, 2.4.2.7, 3.4.1.5, 3.4.1.6, 3.4.1.7, 3.4.1.8, 4.
- Attachment C: [C-] 1.3, 3, 4.
- Attachment D: [D-] 1.1, 2.1.4.4.5, 2.1.5.3, 3.4.
- Attachment E: [E-] 6.2.2.
- Attachment F: [F-] 6.
- Attachment G: [G-] all.

Attachments		Total Number of Pages
Attachment A.	Event Trees	74
Attachment B.	System/Pivotal Event Analysis – Fault Trees	105
Attachment C.	Active Component Reliability Data Analysis	59
Attachment D.	Passive Equipment Failure Analysis	99
Attachment E.	Human Reliability Analysis	76
Attachment F.	Fire Analysis	34
Attachment G.	Event Sequence Quantification Summary Tables	100
Attachment H.	Excel Spreadsheet, SAPHIRE Model, and Supporting Files	4 + CD

RECORD OF REVISIONS

9. No.	10. Reason For Revision	11. Total # of Pgs.	12. Last Pg. #	13. Originator (Print/Sign/Date)	14. Checker (Print/Sign/Date)	15. EGS (Print/Sign/Date)	16. Approved/Accepted (Print/Sign/Date)
00A	Initial issue	619	H-2	Dawn Martin-Miller/ See page 2	See Page 3	M. Frank	M. Wisenburg
00B	The purpose of this revision is to respond to the condition reports noted in Box 8. This revision does not update the analysis to design changes that may have occurred since Rev. 00A was approved.	763	H-4	Laura Plumb 11/18/08	Robert Garrett 11/18/08	Michael Frank 11/19/08	Michael Frank 1/9/09

DISCLAIMER

The analysis contained in this document was developed by Bechtel SAIC Company, LLC (BSC) and is intended solely for the use of BSC in its work for the Yucca Mountain Project (YMP).

Continuation of Item 8 Notes/Comments from page 1

CR Action	Location of Change	Description of Change
11966-005	Pages [C] 25, 46	Revision of the determination of the jib crane drop rate is reflected though the text and revision of the drop rate figure.
11989-005	Page 15	Text of the bullet beginning with "Intentional malevolent…" has been modified as specified in the CR.
12002-005	Pages [C-] 25-26	The total plant ratio used to develop crane drop estimates was changed to 43/109 as recommended. The affected drop rates were updated in response.
12101-004	Pages [B2]- 2, 21, and 22	Site Transporter text was updated as specified in the CR.
12103-009 12103-010	Pages 143-148, [B1-] 20-21, 27-29, [B2-] 12, 23-25, 28-29, [B3-] 4-9, 11, 13-14 [B4-] 2-3, 7, 11-12	SAPHIRE calculation type defaults were reflected in numerous tables, accompanied in many cases by superscripts to enhance clarity. Inconsistencies between SAPHIRE and the documentation were corrected (to include replacement of figures, graphics, and tables).
12105-005	Pages 143-148, 153, 161, 180, [A] 12-16, [B1-] 23-24, 31-32, [B2-] 14, 27-29, [B3-] 7-9, [C-] 44-54, [D-] 14, 51-56, 71, 86, [E-] 55-57, [F-] 27, [G-] 5-6, 8, 15, 26-28, 36-38, 46, 48, 50-52, 56-57, 70	Blank table cells were amended by either merging cells, entering the correct text, entering an em dash, graying cell lines, or adding a footnote to the table.
12176-005	Page 197	The final three sentences preceding Table 6.9-1 were modified to increase the clarity of the table.
12245	Pages 134, 141-142, [B3-] 4-9, 11, 13-14, [C-] 37-42, [G-] 4, 6-7, 9-14, 16-25, 27-35, 58-70, 78, 80-85	CR 12245 called for more robust common-cause failure (CCF) probability distributions. The more robust distributions were developed and incorporated in the SAPHIRE model. These changes required text and figure updates throughout the document.
12310	Pages 143-148, [B1-] 23-24, 31-32, [B2-] 12, 23-25, [B3-] 4-9, 11, 13-14, [B4-] 2-3, 7, 11-12	It was ensured through actions taken in response to CR 12103 and 12245 that SAPHIRE is consistent with the documentation.

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ACRONYMS AND ABBREVIATIONS

Acronyms

BOP	balance of plant
BSC	Bechtel SAIC Company, LLC
CCF	common-cause failure
CRCF	Canister Receipt and Closure Facility
CTM	canister transfer machine
DOE	U.S. Department of Energy
DPC	dual-purpose canister
EFC	error-forcing context
ESD	event sequence diagram
FEA	finite element analysis
FEM	finite element modeling
FTA	fault tree analysis
GROA	geologic repository operations area
HAM	horizontal aging module
HAZOP	hazard and operability
HDPC	horizontal dual-purpose canister
HEPA	high-efficiency particulate air
HFE	human failure event
HIC	high-integrity canister
HLW	high-level radioactive waste
HRA	human reliability analysis
HSTC	horizontal shielded transfer cask
HTC	a transportation cask that is never upended
HVAC	heating, ventilation, and air conditioning
IET	initiator event tree
IHF	Initial Handling Facility
ITC	important to criticality
ITS	important to safety
LLNL	Lawrence Livermore National Laboratory
LLW	low-level radioactive waste
LLWF	Low-Level Waste Facility
LOS	loss of shielding
MLD	master logic diagram

ACRONYMS AND ABBREVIATIONS (Continued)

NAICS	North American Industry Classification System
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
PCSA	preclosure safety analysis
PDF	probability density function
PEFA	passive equipment failure analysis
PFD	process flow diagram
PLC	programmable logic controller
PRA	probabilistic risk assessment
PSC	procedural safety controls
PSF	performance-shaping factor
RF	Receipt Facility
SFTM	spent nuclear fuel transfer machine
SLS	steel/lead/steel
SNF	spent nuclear fuel
SPM	site prime mover
SPMRC	site prime mover railcar
SPMTT	site prime mover truck trailer
SRET	system response event tree
SSC	structure, system, or component
SSCs	structures, systems, and components
TAD	transportation, aging, and disposal
TEV	transport and emplacement vehicle
TNT	trinitrotoluene
WHF	Wet Handling Facility
WPTT	waste package transfer trolley
YMP	Yucca Mountain Project

ACRONYMS AND ABBREVIATIONS (Continued)

Abbreviations

AC	alternating current
BTU	British Thermal Unit
°C	degree Celsius
ft	foot, feet
ft^3	cubic foot
hr	hour
K	Kelvin
k_{eff}	effective neutron multiplication factor
kPa	kilopascal
kV	kilovolt
lb	pound
lb-mol	pound-mole
m^2	square meter
psi	pound per square inch
°R	degree Rankine
V	volt
yr	year

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1. PURPOSE

This document and its companion document entitled *Intra-Site Operations and BOP Event Sequence Development Analysis* (Ref. 2.2.29) constitute a portion of the preclosure safety analysis (PCSA) that is described in its entirety in the safety analysis report that will be submitted to the U.S. Nuclear Regulatory Commission (NRC) as part of the Yucca Mountain Project (YMP) license application. These documents are part of a collection of analysis reports that encompass all waste handling activities and facilities of the geologic repository operations area (GROA) from the beginning of operation to the end of the preclosure period. The *Intra-Site Operations and BOP Event Sequence Development Analysis* (Ref. 2.2.29) describes the identification of initiating events and the development of potential event sequences that emanate from them. This analysis uses the resulting event sequences to perform a quantitative analysis of the event sequences for the purpose of categorization per the definition provided by 10 CFR 63.2 (Ref. 2.3.2).

The PCSA uses probabilistic risk assessment (PRA) technology derived from both nuclear power plant and aerospace methods and applications in order to perform analyses to comply with the risk-informed aspects of 10 CFR 63.111 and 63.112 (Ref. 2.3.2) and to be responsive to the acceptance criteria articulated in the *Yucca Mountain Review Plan, Final Report* (Ref. 2.2.69). The PCSA, however, limits the use of PRA technology to identification and development of event sequences that might lead to direct exposure of workers or onsite members of the public; radiological releases that may affect the workers or public (onsite and offsite) and criticality.

The radiological consequence assessment relies on bounding inputs with deterministic methods to obtain bounding dose estimates. These were developed using broad categories of scenarios that might cause a radiological release or direct exposure to workers and the public, both onsite and offsite. These broad categories of scenarios were characterized by conservative meteorology and dispersion parameters, conservative estimates of material at risk, conservative source terms, conservative leak path factors, and filtration of releases via facility high-efficiency particulate air (HEPA) filters, when applicable. After completion of the event sequence development and categorization in this analysis, each Category 1 and Category 2 event sequence was conservatively matched with one of the categories of dose estimates. The event sequence analyses also serve as input to the PCSA criticality analyses by identifying the event sequences and end states where conditions leading to criticality are in Category 1 or 2.

An event sequence is defined in 10 CFR 63.2 (Ref. 2.3.2) as follows:

> "A series of actions and/or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel. Those event sequences that are expected to occur one or more times before permanent closure of the geologic repository operations area are referred to as Category 1 event sequences. Other event sequences that have at least one

chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences."

As an extrapolation of the definition of Category 2 event sequences, sequences that have less than one chance in 10,000 of occurring before permanent closure are identified as beyond Category 2. Consequence analyses are not required for those event sequences.

10 CFR 63.112, Paragraph (e) and Subparagraph (e)(6) (Ref. 2.3.2) require analyses to identify the controls that are relied upon to limit or prevent potential event sequences or mitigate their consequences. Subparagraph (e)(6) specifically notes that the analyses should include consideration of "means to prevent and control criticality." The PCSA criticality analyses employ specialized deterministic methods that are beyond the scope of the present analysis. However, the event sequence analyses serve as an input to the PCSA criticality analyses by identifying the event sequences and end states where conditions leading to criticality are in Category 1 or 2. Some event sequence end states include the phrase "important to criticality" (ITC). This indicates that the event sequence has a potential for reactivity increase that should be analyzed to determine if reactivity can exceed the upper subcriticality limit.

In order to determine the criticality potential for each waste form and associated facility and handling operations, criticality sensitivity calculations are performed. These calculations evaluate the impact on system reactivity to variations in each of the parameters ITC during the preclosure period. The parameters are waste form characteristics, reflection, interaction, neutron absorbers (fixed and soluble), geometry, and moderation. The criticality sensitivity calculations determine the sensitivity of the effective neutron multiplication factor (k_{eff}) to variations in any of these parameters as a function of the other parameters. The PCSA criticality analyses determined the parameters that this event sequence analysis should include. The presence of a moderator in association with a path to exposed fuel was required to be explicitly modeled in the event sequence analysis because such events could not be deterministically found to be incapable of exceeding the upper subcriticality limit. Other situations treated in the event sequence analysis for similar reasons are multiple U.S. Department of Energy (DOE) spent nuclear fuel (SNF) canisters in the Canister Receipt and Closure Facility (CRCF) in the same general location, and presence of sufficient soluble boron in the pool in the Wet Handling Facility (WHF).

The initiating events considered in the PCSA define what could occur within the site GROA and are limited to those that constitute a hazard to a waste form while it is present in the GROA; i.e., an internal event due to a waste processing operation conducted in the GROA, or an external event that imposes a potential hazard to a waste form, or waste processing systems, or personnel (e.g., seismic or wind energy, flood waters). Such initiating events are included when developing event sequences for the PCSA. However, initiating events that are associated with conditions introduced in structures, systems, and components (SSCs) before they reach the site (e.g., drops of casks, canisters, or fuel assemblies during loading at a reactor site; improper drying, closing, or inerting at the reactor site; rail accidents during transport; tornado or missile strikes on a transportation cask) or nonconformances during cask or canister manufacture (i.e., resulting in a reduction of containment strength) are not within the scope of the PCSA. Such potential precursors are subject to deterministic regulations (e.g., 10 CFR Part 50 (Ref. 2.3.1), 10 CFR Part 71 (Ref. 2.3.3), and 10 CFR Part 72 (Ref. 2.3.4)) and associated quality

assurance programs. As a result of compliance to such regulations, the SSCs are deemed to pose no undue risk to health and safety. Although the analyses do not address quantitative probabilities, it is clear that the use of conservative design criteria and the implementation of quality assurance controls will result in unlikely exposures to radiation.

Other boundary conditions used in the PCSA include:

- Plant operational state. The initial state of the facility is normal with each system operating within its vendor-prescribed operating conditions.

- No other simultaneous initiating events. It is standard practice to not consider the occurrence of other initiating events (human-induced or naturally occurring) during the time span of an event sequence because (a) the probability of two simultaneous initiating events within the time window is small and, (b) each initiating event will cause operations to be terminated, which further reduces the conditional probability of the occurrence of a second initiating event, given that the first has occurred.

- Equipment failure mode. The failure mode of a structure, system, or component (SSC) corresponds to that required to make the initiating or pivotal event occur.

- Fundamental to the basis for the use of industry-wide reliability parameters within the PCSA, such as failure rates, is the use of structures, systems, and components (SSCs) within the GROA that conform to NRC accepted consensus codes and standards, and other regulatory guidance.

- Intentional malevolent acts, such as sabotage and other security threats, were considered in a separate safeguards and security analysis performed by others.

As stated, the scope of the PCSA is limited to internal initiating events originating within the GROA boundary and external initiating events that have their origin outside the GROA boundary, but can affect buildings and/or equipment within the GROA. External event analyses are documented in other PCSA documents (e.g., *External Events Hazards Screening Analysis* (Ref. 2.2.26)). Internal event identification (using a master logic diagram (MLD) and hazard and operability (HAZOP) evaluation), event sequence development and grouping, and related facility details are provided in *Intra-Site Operations and BOP Event Sequence Development Analysis* (Ref. 2.2.29), which also documents the methodology and process employed and initiates the analysis that is completed here.

This document uses event trees from the event sequence development analysis (Ref. 2.2.29) to quantify the event sequences for each waste form. Quantification refers to the process of obtaining the mean frequency of each event sequence for the purpose of categorization. This document shows the categorization of each event sequence based on:

- Mean frequency associated with the event sequence frequency distribution.

- Uncertainty associated with the event sequence frequency distributions.

- Material at risk for each Category 1 and Category 2 event sequence for purposes of dose calculations.
- Important to safety (ITS) SSCs.
- Compliance with the nuclear safety design bases.
- Procedural safety controls (PSCs) required for operations.

Other PCSA documents, which are not referenced here, cover the reliability and categorization of external events and summarize PSCs and nuclear safety design bases.

2. REFERENCES

2.1 PROJECT PROCEDURES/DIRECTIVES

2.1.1 EG-PRO-3DP-G04B-00037, REV 14. *Calculations and Analyses*. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20081114.0008.

2.1.2 EG-PRO-3DP-G04B-00046, Rev. 13. *Engineering Drawings*. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG. 20081114.0009.

2.1.3 IT-PRO-0011, REV 10. *Software Management*. Las Vegas, Nevada: Bechtel SAIC Company. ACC: DOC.20080923.0003.

2.1.4 LS-PRO-0201, REV 7. *Preclosure Safety Analysis Process*. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20081119.0007.

2.2 DESIGN INPUTS

This PCSA is based on a snapshot of the design. The reference design documents are appropriately documented as design input in this section. Since the safety analysis is based on a snapshot of the design, referencing subsequent revisions to the design documents as described in EG-PRO-3DP-G04B-00037, *Calculations and Analyses* (Ref. 2.1.1, Section 3.2.2.F) that implement PCSA requirements flowing from the safety analysis would not be appropriate for the purpose of this document. There are no superseded or cancelled documents associated with the modifications that led to the issuance of this revision. Cancelled or superseded documents associated with the portions of this document for which the snapshot has not yet been updated are designated herein with a dagger (†).

Design Inputs are listed in this section and the Attachment sections listed in Section 2.5.

The inputs in this section noted with an asterisk (*) indicate that they fall into one of the designated categories described in Section 4.1, relative to suitability for intended use.

2.2.1 *Ahrens, M. 2000. *Fires in or at Industrial Chemical, Hazardous Chemical and Plastic Manufacturing Facilities, 1988-1997 Unallocated Annual Averages and Narratives*. Quincy, Massachusetts: National Fire Protection Association. TIC: 259997.

2.2.2 AIChE (American Institute of Chemical Engineers) 2000. *Guidelines for Chemical Process Quantitative Risk Analysis*. 2nd Edition. New York, New York: Center for Chemical Process Safety of the American Institute of Chemical Engineers. TIC: 251253. ISBN: 0-8169-0720-X. TIC: 251253.

2.2.3 *ANSI/ANS-58.23-2007. *Fire PRA Methodology*. La Grange Park, Illinois: American Nuclear Society. TIC: 259894.

2.2.4 *Apostolakis, G. and Kaplan, S. 1981. "Pitfalls in Risk Calculations." *Reliability Engineering, 2*, 135-145. Barking, England: Applied Science Publishers. TIC: 253648.

2.2.5 ASCE/SEI 7-05. 2006. *Minimum Design Loads for Buildings and Other Structures*. Including Supplement No. 1. Reston, Virginia: American Society of Civil Engineers. TIC: 258057. ISBN: 0-7844-0809-2.

2.2.6 ASME (American Society of Mechanical Engineers) RA-S-2002. *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Application*s. New York, New York: American Society of Mechanical Engineers. TIC: 255508. ISBN: 0-7918-2745-3.

2.2.7 ASME 2004. *2004 ASME Boiler and Pressure Vessel Code*. 2004 Edition. New York, New York: American Society of Mechanical Engineers. TIC: 256479.

2.2.8 *Atwood, C.L.; LaChance, J.L.; Martz, H.F.; Anderson, D.J.; Englehardt, M.; Whitehead, D.; and Wheeler, T. 2003. *Handbook of Parameter Estimation for Probabilistic Risk Assessment*. NUREG/CR-6823. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20060126.0121.

2.2.9 *Battelle 2001. *Comparative Risks of Hazardous Materials and Non-Hazardous Materials Truck Shipment Accidents/Incidents, Final Repor*t. Washington, D.C.: U.S. Department of Transportation, Federal Motor Carrier Safety Administration. ACC: MOL.20080228.0002.

2.2.10 *Benhardt, H.C.; Eide, S.A.; Held, J.E.; Olsen, L.M.; and Vail, R.E. 1994. *Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities (U)*. WSRC-TR-93-581. Aiken, South Carolina: Westinghouse Savannah River Company, Savannah River Site. ACC: MOL.20061201.0160.

2.2.11 Bjorkman, G.; Chuang, T-J.; Einziger, R.; Malik, S.; Malliakos, A.; Mitchell, J.; Navarro, C.; Ryder, C.; Shaukat, S.; Ulses, A.; and Zigh, G. 2007. *A Pilot Probabilistic Risk Assessment Of a Dry Cask Storage System at a Nuclear Power Plant*. NUREG-1864. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20070913.0518.

2.2.12 BSC (Bechtel SAIC Company) 2005. *Thermal Performance of Spent Nuclear Fuel During Dry Air Transfer-Initial Calculations*. 000-00C-DSU0-03900-000-00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20050110.0003.

2.2.13 *BSC 2006. *Low-Level Waste Management Plan*. 000-30R-MW00-00100-000 REV 000. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20061218.0016.

2.2.14 BSC 2007. *Aging Facility Cask Tractor Mechanical Equipment Envelope*. 170-MJ0-HAT0-00601-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070910.0016.

2.2.15 BSC 2007. *Aging Facility Cask Transfer Trailers Mechanical Equipment Envelope*. 170-MJ0-HAT0-00201-000 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070518.0002.

2.2.16 †BSC 2007. *Basis of Design for the TAD Canister-Based Repository Design Concept*. 000-3DR-MGR0-00300-000-001. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071002.0042.

2.2.17 †BSC 2007. *Cask Transfer Trolley and Site Transporter Sliding/Rocking Calculation*. 000-MJC-H000-00200-000 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0010.

2.2.18 *BSC 2007. *Diesel Fuel Oil Storage Diesel Fuel Oil System Piping and Instrumentation Diagram*. 70A-M60-PSO0-00103-000 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070904.0003.

2.2.19 BSC 2007. *GROA External Dose Rate Calculation.* 000-PSA-MGR0-01300-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071023.0003.

2.2.20 †BSC 2007. *Mechanical Handling Design Report-Site Transporter*. 170-30R-HAT0-00100-000-000. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071217.0015.

2.2.21 BSC 2007. *Naval Long Waste Package Vertical Impact on Emplacement Pallet and Invert*. 000-00C-DNF0-00100-000-00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071017.0001.

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2.3 DESIGN CONSTRAINTS

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2.3.2 10 CFR 63. 2007. Energy: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada. U.S. Nuclear Regulatory Commission.

2.3.3 10 CFR 71. 2007. Energy: Packaging and Transportation of Radioactive Material. U.S. Nuclear Regulatory Commission. ACC: MOL.20070829.0114.

2.3.4 10 CFR 72. 2007. Energy: Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste. U.S. Nuclear Regulatory Commission.

2.4 DESIGN OUTPUTS

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2.5 ATTACHMENT REFERENCES

Attachment A: references are cited in Section 2.2 of main body

Attachment B: references are cited in Section B1.1, B2.1, and B3.1

Attachment C: references are cited in Section C5

Attachment D: references are cited in Section D4

Attachment E: references are cited in Section E8

Attachment F: references are cited in Section F2

Attachment G: references are cited in Sections 2.2 and 2.4 of main body

Attachment H: none

3. ASSUMPTIONS

3.1 ASSUMPTIONS REQUIRING VERIFICATION

None used.

3.2 ASSUMPTIONS NOT REQUIRING VERIFICATION

3.2.1 General Analysis Assumptions

Assumption–Equipment and SSCs designed and purchased for the Yucca Mountain repository are of the population of equipment and SSCs represented in United States industry-wide reliability information sources. Furthermore, the uncertainty in reliability is represented by the variability of reliabilities across this population.

Rationale–Although the repository features some unique pieces of equipment at the system level (e.g., the waste package transfer trolley (WPTT) and the cask transfer trolley used in the waste handling facilities), at the component level the repository relies on proven and established technologies. The industry-wide information sources include historical reliability information at the component level. Such experience is relevant to the repository because the repository relies on components similar to the ones represented in the information sources. In some cases, system-level information, such as crane load drop rates, from the industry-wide information sources are used. It is appropriate to use such information because it represents similar pieces of equipment at the system level. In addition, drawing from a wide spectrum of sources takes advantage of many observations, which yields better statistical information regarding the uncertainty associated with the resulting reliability estimates.

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4. METHODOLOGY

4.1 QUALITY ASSURANCE

This analysis has been prepared in accordance with *Calculations and Analyses* (Ref. 2.1.1) and *Preclosure Safety Analysis Process* (Ref. 2.1.4). Therefore, the approved version is designated as "QA: QA."

In general, input designated "QA: QA" is used; however, some of the inputs that are cited are designated "QA: N/A." The suitability of these inputs for the intended use is justified as follows.

Documentation of suitability for intended use of "QA: N/A" drawings: Engineering drawings are prepared using the "QA: QA" procedure *Engineering Drawings* (Ref. 2.1.2). They are checked by an independent checker and reviewed for constructability and coordination before review and approval by the engineering group supervisor and the discipline engineering manager (Ref. 2.1.2, Section 3.2.2, and Attachments 3 and 5). The check, review, and approval process provides assurance that these drawings accurately document the design and operational philosophy of the facility. For this reason, they are suitable for their intended use as sources of input to this analysis.

Documentation of suitability for intended use of "QA: N/A" engineering calculations or analyses: Engineering calculations and analyses are prepared using the "QA: QA" procedure *Calculations and Analyses* (Ref. 2.1.1). They are checked by an independent checker and reviewed for coordination before review and approval by the engineering group supervisor and the discipline engineering manager. The check, review, and approval process provides assurance that these calculations and analyses accurately document the design and operation of the facility. For this reason, they are suitable for their intended use as sources of input to this analysis.

Documentation of suitability for intended use of engineering studies (which are "QA: N/A"): In a few instances, studies are used as inputs to this analysis. The uses of inputs from studies are made clear by the context of the discussion at the point of use. The use of studies is acceptable for committed analyses, such as the present analysis, provided that the results are not used for procurement, fabrication, or construction purposes. Because the present analysis is not used for procurement, fabrication, or construction purposes, the use of studies is acceptable. Therefore, the studies that are used as inputs are suitable for their intended uses.

Documentation of suitability for intended use of BSC design guides (which are "QA: N/A"): The uses of inputs from design guides are made clear by the context of the discussion at the point of use. Design guides are used as inputs only when specific design documents, such as drawings, calculations, and design reports are not available at the present level of design development. Therefore, the design guides that are used as inputs are suitable for their intended uses.

Documentation of suitability for intended use of BSC engineering standards (which are "QA: N/A"): Engineering standards are used in this analysis as the basis for the numbering system for basic events. The uses of inputs from BSC engineering standards are made clear by the context of the discussion at the point of use. Therefore, the design guides that are used as inputs are suitable for their intended uses.

Documentation of suitability for intended use of BSC Interoffice memorandum: Due to the early nature of the design of some systems, the only available sources for the information used are interoffice memorandum. The information used from these sources are conservative estimates and appropriate for their intended use.

Documentation of suitability for intended use of inputs from outside sources: The uses of inputs from outside sources are made clear by the context of the discussion at the point of use. These uses fall into the following categories and are justified as follows (in addition to the justifications provided at the point of use):

1. Some inputs are cited as sources of the methods used in the analysis. These inputs are suitable for their intended uses because they represent commonly accepted methods of analysis among safety analysis practitioners or, more generally, among scientific and engineering professionals.

2. Some inputs are cited as examples of applications of analytical methods by others. These inputs are suitable for their intended uses because they illustrate applicable methods of analysis.

3. Some inputs are cited as sources of historical safety-related data. These inputs are suitable for their intended uses because they represent historical data that is commonly accepted among safety analysis practitioners.

4. Some inputs are cited as sources of accepted practices as recommended by codes, standards, or review plans. These inputs are suitable for their intended uses because they represent codes, standards, or review plans that are commonly accepted by practitioners of the affected professional disciplines.

5. Some inputs provide information specific to the Yucca Mountain Repository that was produced by organizations other than BSC. These inputs are suitable for their intended uses because they provide information that was developed for the Yucca Mountain Repository under procedures that apply to the organization that produced the information.

4.2 USE OF SOFTWARE

4.2.1 Level 1 Software

This section addresses software used in this analysis as Level 1 software, as defined in *Software Management* (Ref. 2.1.3, Attachment 12). SAPHIRE V. 7.26 STN 10325-7.26-01 (Ref. 2.2.76) is used in this analysis for PRA simulation and analyses. The SAPHIRE software is used on a personal computer running Windows XP inside a VMware virtual machine; it is also listed in the