RAI: Volume 3, Chapter 2.2.1.2.2, Number 2:

Clearly state why the sum of the means of the probability distributions for the respective undetected defects that could lead to early failure do not equal the mean of the overall probability distributions for undetected defects in the waste package and drip shield (i.e., the early failure probabilities).

Basis: In SAR, Sections 2.3.6.6.3.2.1-2.3.6.6.3.2.6, DOE provides the mean values of the probability distributions for the six events that could lead to early failure of the waste package. The sum of the means of the probability distributions for these six events does not equal the mean value for the overall probability of an undetected error in the waste package (i.e., the early failure probability), which is given in SAR, Section 2.3.6.6.3.2.7. The same statement is true for the drip shield (where there are four events that could lead to early failure), referencing SAR, Sections 2.3.6.8.4.3.2.1-2.3.6.8.4.3.2.5.

1. RESPONSE

Investigation has determined that the mean probabilities for the event sequences leading to early failure of the waste package, as reported in SAR Sections 2.3.6.6.3.2.1 to 2.3.6.6.3.2.6, are incorrect. Similarly, the mean probabilities for the event sequences leading to early failure of the drip shield, as reported in SAR Sections 2.3.6.8.4.3.2.1 to 2.3.6.8.4.3.2.4, are also incorrect. These values are intermediate outputs from a fault tree analysis conducted with the SAPHIRE software. The SAPHIRE file was not properly configured to correctly display the desired intermediate values. The configuration error has been corrected in the SAPHIRE files so that the intermediate results are correctly displayed. The configuration error only affects display of results and does not affect the results of the SAPHIRE calculations; thus, the overall probabilities of early failure of waste packages and drip shields are unchanged.

Table 1 displays the correct values in the same format as Table 6-8 of *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007), the source of the values presented in the SAR sections listed above. DOE will amend ANL-EBS-MD-000076, “Analysis of Mechanisms for Early Waste Package/Drip Shield Failure,” to correct Table 6-8 in that document.

Table 1. Parameters from Scenario End-State Uncertainty Distributions for Undetected Defects

Event Sequence	Mean Value	Median Value
Base metal flaw (waste package outer corrosion barrier)	1.25×10^{-7}	7.96×10^{-8}
Base metal flaw (drip shield)	1.25×10^{-7}	7.96×10^{-8}
Improper heat treatment (waste package outer corrosion barrier)	3.73×10^{-5}	1.42×10^{-5}
Improper heat treatment (waste package outer corrosion barrier lid)	3.50×10^{-5}	1.35×10^{-5}
Improper heat treatment (drip shield)	1.97×10^{-6}	1.42×10^{-7}
Weld filler flaws (waste package outer corrosion barrier)	1.25×10^{-7}	7.96×10^{-8}
Weld filler flaws (drip shield)	1.25×10^{-7}	7.96×10^{-8}
Emplacement error (drip shield)	2.19×10^{-9}	3.40×10^{-10}
Handling error (waste package outer corrosion barrier)	9.71×10^{-7}	2.86×10^{-7}
Low-plasticity burnishing (waste package outer corrosion barrier)	3.77×10^{-5}	7.27×10^{-6}

The values provided in Table 1 are tabulated from the Monte Carlo analysis conducted with SAPHIRE. Using the values presented in Table 1, the sum of the mean probabilities for the event sequences leading to waste package early failure is 1.11×10^{-4}, and the sum of the mean probabilities for event sequences leading to drip shield early failure is 2.22×10^{-6}. The overall mean probability for waste package early failure is given as 1.13×10^{-4} (SAR Section 2.3.6.6.3.2.7). The overall mean probability for drip shield early failure is given as 2.21×10^{-6} (SAR Section 2.3.6.8.4.3.2.5). These overall mean probabilities are the means of lognormal distributions that are fitted to the ensemble of overall early failure probabilities that are generated by the Monte Carlo analysis. Hence, the results are slightly different between the sum of the mean probabilities listed in Table 1, and the overall mean probability determined from the lognormal distribution fitted to the SAPHIRE results.

2. COMMITMENTS TO NRC

DOE commits to update the LA as described in Section 3 below. The change to be made to the LA will be included in a future LA update.

3. DESCRIPTION OF PROPOSED LA CHANGE

In each of SAR Sections 2.3.6.6.3.2.1 to 2.3.6.6.3.2.6, and in SAR Sections 2.3.6.8.4.3.2.1 to 2.3.6.8.4.3.2.4, DOE will replace the mean and median probabilities provided for each event sequence with the corrected values displayed in Table 1 in a future LA update.

4. REFERENCES

SNL (Sandia National Laboratories) 2007. *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure*. ANL-EBS-MD-000076 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070629.0002; DOC.20071003.0015; LLR.20080311.0094; DOC.20080918.0002.

RAI: Volume 3, Chapter 2.2.1.2.2, Number 3

Justify use of the following data, given in SNL (2007a), to calculate the early failure probabilities:

(a) Use of the value from Swain and Guttmann (1983, Item 2, Table 20-10) for the events

(i) CHECK_BM_FLAW in the event tree for evaluating improper base metal selection for the waste package outer corrosion barrier

(ii) WELD_FILLER_ISP-WP in the event tree for evaluating weld filler material defects in the waste package outer corrosion barrier

(iii) CHECK_BM_FLAW_DS in the event tree for evaluating improper base metal selection for the drip shield

(iv) WELD_FILLER_ISP-DS in the event tree for evaluating weld filler material defects in the drip shield

Basis: These events are recovery actions that correct the failure of the prior event in the respective sequences. It may be more appropriate to use Swain and Guttmann (1983, Table 20-22). The events are used for the probability calculations found in SAR, Sections 2.3.6.6.3.2.1-2.3.6.6.3.2.6 for waste package early failure probability and SAR Sections 2.3.6.8.4.3.2.1-2.3.6.8.4.3.2.4 for drip shield early failure probability.

(b) Use of the low mean values from Benhardt et al., (1994) for the events

(i) HT_OPERATOR_ERROR in the event tree for evaluating improper heat treatment of the waste package outer corrosion barrier;

(ii) SNORKEL_ATTACHMENT_FAIL in the event tree for evaluating improper heat treatment of the waste package outer corrosion barrier

(iii) HT_OPERATOR_ERROR in the event tree for evaluating improper heat treatment of the waste package outer corrosion barrier lid

(iv) WP-LPB-OPERATOR in the event tree for evaluating low-plasticity burnishing treatment of the waste package outer corrosion barrier lid

(v) HT_DS in the event tree for evaluating improper heat treatment of the drip shield

(vi) HT_OPERATOR_ERROR in the event tree for evaluating improper heat treatment of the drip shield

(vii) DS_EMP_ANN in the event tree for evaluating improper emplacement of the drip shield

Basis: Because DOE has not provided a detailed analysis of factors such as equipment design, operating environment, and operator training, it may be more appropriate to use nominal mean values from Benhardt et al., (1994). DOE should justify the use of low mean values. The events are used for the probability calculations found in SAR, Sections 2.3.6.6.3.2.1-2.3.6.6.3.2.6 for waste package early failure probability and SAR, Sections 2.3.6.8.4.3.2.1-2.3.6.8.4.3.2.4 for drip shield early failure probability.

1. RESPONSE

Waste packages and drip shields are identified as important to waste isolation (SAR Table 1.9-8) and will have stringent controls on their fabrication and handling (SAR Table 1.9-9, drip shield and waste package design control parameters).

As discussed in SAR Section 1.5.2, the waste package outer corrosion barrier is specifically designed as a corrosion barrier, not a pressure vessel. However, it is constructed in accordance with the applicable technical requirements of *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components, including material, fabrication, and examination requirements and selected administrative requirements. The outer corrosion barrier is evaluated both against stress limits, consistent with the design limits specified in *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001), and against an energy absorption failure measure for margin analysis.

As discussed in SAR Tables 1.3.4-5 and 1.9-9, design control parameter 07-09, the drip shield fabrication specification will require that procedures developed by the fabricator be developed consistent with standard nuclear industry practices, for the applicable codes and standards listed in SAR Table 1.3.2-5. The drip shield shall be fabricated in accordance with the requirements of *ASME Boiler and Pressure Vessel Code* (ASME 2001 Section III, Division 1, Subsection NC (Class 2 pressure vessel)). It shall be inspected by an Authorized Nuclear Inspector and certified as to meeting the specific provisions of *ASME Boiler and Pressure Vessel Code* as identified in *Yucca Mountain Project Engineering Specification Prototype Drip Shield* (BSC 2007, Section 1.4).

1.1 PART (A) JUSTIFICATION FOR USE OF VALUES FROM SWAIN AND GUTTMANN (1983, ITEM 2, TABLE 20-10)

1.1.1 Justification of Values Used for CHECK_BM_FLAW

Although the CHECK_BM_FLAW top event includes the word “CHECK” in its label, the event is not representing a check in the sense of that relevant to Table 20-22 in Swain and Guttmann (1983) (i.e., a checker detecting errors made by others). Rather, the event refers to a measurement of the chemical composition of the waste package base metal by the material supplier above and beyond that done in support of preparing a material certification. Such an inspection is consistent with SAR Tables 1.5.2-7 and 1.9-9 (postclosure (design) control parameter 03-19) and in compliance with 10 CFR 63.142. This inspection might be accomplished with an X-Ray spectrometer (SNL 2007, Section 6.3.2) or other instrument. Since this confirmation is performed by an independent organization and procedure, it is appropriately represented by a human error probability of improperly reading and recording a digital display (i.e., a nominal (median) probability of 0.001 and an error factor of 3) (Swain and Guttmann 1983, Table 20-10).

Because the fabrication and handling of the waste package outer corrosion barrier will be accomplished under stringent controls (SAR Table 1.9-9, waste package design control parameters) and in accordance with standard nuclear industry practices, requirements, and procedures (e.g., *ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components), it is expected that the material supplier and/or fabricator would check the results of the measurement of the chemical composition of the waste package base metal (accomplished in support of preparing a material certification) and the Yucca Mountain Project would check the material certifications and the fabricator's records upon receipt. Thus, the probability of the base metal being acceptable (BM_FLAW) upon receipt by the Yucca Mountain Project is expected to be higher than is being represented in the event tree. For these reasons, the treatment is appropriate and justified.

1.1.2 Justification of Values Used for WELD_FILLER_ISP-WP

The WELD_FILLER_ISP-WP top event is not representing a check in the sense of that relevant to Table 20-22 in Swain and Guttmann (1983) (i.e., a checker detecting errors made by others). Rather, the event refers to a measurement of the chemical composition of the weld filler material above and beyond that done by the material supplier in support of preparing a material certification. Such an inspection is consistent with SAR Tables 1.5.2-7 and 1.9-9 (postclosure (design) control parameter 03-14). This inspection might be accomplished with an X-Ray spectrometer (SNL 2007, Section 6.3.2) or other instrument. Since this confirmation is performed by an independent organization and procedure, it is appropriately represented by a human error probability of improperly reading and recording a digital display (i.e., a nominal (median) probability of 0.001 and an error factor of 3) (Swain and Guttmann 1983, Table 20-10).

Because the fabrication and handling of the waste package outer corrosion barrier will be accomplished under stringent controls (SAR Table 1.9-9, waste package design control parameters) and in accordance with standard nuclear industry practices, requirements, and procedures (e.g., *ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components), it is expected that the weld filler material supplier and/or the fabricator would check the results of the measurement of the chemical composition of the weld filler metal (accomplished in support of preparing a material certification), the Yucca Mountain Project (in the case of the final closure weld) would check the weld filler material certification upon receipt, and the Yucca Mountain Project would check the fabricator's records upon receipt. Because the results of the inspection associated with the WELD_FILLER_ISP-WP top event would be checked, further reducing the probability of failure to properly detect the use of improper weld filler material, the treatment is appropriate and justified.

1.1.3 Justification of Values Used for CHECK_BM_FLAW_DS

Although the CHECK_BM_FLAW_DS top event includes the word "CHECK" in its label, the event is not representing a check in the sense of that relevant to Table 20-22 in Swain and Guttmann (1983) (i.e., a checker detecting errors made by others). Rather, the event refers to a measurement of the chemical composition of the drip shield base metal above and beyond that done by the material supplier in support of preparing a material certification. Such an inspection is consistent with SAR Tables 1.3.4-5 and 1.9-9 (postclosure (design) control parameter 07-09). This inspection might be accomplished with an X-Ray spectrometer (SNL 2007, Section 6.3.2) or other instrument. Since this confirmation is performed by an independent organization and procedure, it is appropriately represented by a human error probability of improperly reading and recording a digital display (i.e., a nominal (median) probability of 0.001 and an error factor of 3) (Swain and Guttmann 1983, Table 20-10).

Because the fabrication and handling of the drip shield will be accomplished under stringent controls (SAR Table 1.9-9, drip shield design control parameters) and in accordance with standard nuclear industry practices, requirements, and procedures (e.g., *ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components), it is expected that the material supplier and/or fabricator would check the results of the measurement of the chemical composition of the waste package base metal (accomplished in support of preparing a material certification) and the Yucca Mountain Project would check the material certifications and the fabricator's records upon receipt. Thus, the probability of the base metal being acceptable (BM_FLAW_DS) upon receipt by the Yucca Mountain Project is expected to be higher than is being represented. For these reasons, the treatment is appropriate and justified.

1.1.4 Justification of Values Used for WELD_FILLER_ISP-DS

The WELD_FILLER_ISP-DS top event does not represent a check in the sense of that relevant to Table 20-22 in Swain and Guttmann (1983) (i.e., a checker detecting errors made by others). Rather, the event refers to a measurement of the chemical composition of the weld filler material above and beyond that done by the material supplier in support of preparing a material certification. Such an inspection is consistent with SAR Tables 1.3.4-5 and 1.9-9 (design control parameter 07-09). This inspection might be accomplished with an X-Ray spectrometer (SNL 2007, Section 6.3.2) or other instrument. Since this confirmation is performed by an independent organization and procedure, it is appropriately represented by a human error probability of improperly reading and recording a digital display (i.e., a nominal (median) probability of 0.001 and an error factor of 3) (Swain and Guttmann 1983, Table 20-10).

Because the fabrication and handling of the drip shield will be accomplished under stringent controls (SAR Table 1.9-9, drip shield design control parameters) and in accordance with standard nuclear industry practices, requirements, and procedures (e.g., *ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components), it is expected that the weld filler material supplier and/or fabricator would check the results of the measurement of the chemical composition of the weld filler metal (accomplished in support of preparing a material certification) and the Yucca Mountain Project would check the material certifications and the fabricator's records upon receipt. Because the results of the inspection associated with the WELD_FILLER_ISP-DS top event would be checked, further reducing the probability of failure to properly detect the use of improper weld filler material, the treatment is appropriate and justified.

1.2 PART (B) JUSTIFICATION FOR USE OF THE LOW MEAN VALUES FROM BENHARDT ET AL. (1994)

Use of the low mean and error factor (EF) values for these basic events is appropriate. Benhardt et al. (1994, Table 4) indicate that use of the nominal case values implies good procedures, training, experience, and environment; use of the high values implies poor procedures, training, experience, and environment; and use of the low value implies excellent procedures, training, experience, and environment. Any repository operation is under strict QA requirements derived from 10 CFR 63.142 (SAR Section 1.9; BSC 2008, Section 2). This is reflected in the fact that the waste packages and drip shields are identified as important to waste isolation (SAR Table 1.9-8) and the fabrication and handling of these components will be accomplished under stringent controls (SAR Table 1.9-9, drip shield and waste package design control parameters) and in accordance with standard nuclear industry practices, requirements, and procedures (e.g., *ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC) for Class 2 components). Thus, it is reasonable to assume excellent procedures, training, experience, and environment are used during waste package and drip shield handling and fabrication processes.

The "Failure of an administrative control" human error probability from Benhardt et al. (1994, Table 4) (mean failure probability of 5.0×10^{-4} and EF of 10 stated to be appropriate for "Routine, repetitive circumstances") is used for the (ii) SNORKEL_ATTACHMENT_FAIL event in the event tree for evaluating improper heat treatment of the waste package outer corrosion barrier and the (v) HT_DS event in the event tree for evaluating improper heat treatment of the drip shield.

The "Failure to respond to a compelling signal" human error probability from Benhardt et al. (1994, Table 4) (mean failure probability of 3.0×10^{-3} and EF of 10 stated to be appropriate for "Few competing signals") is used for the HT_OPERATOR_ERROR event in the event trees for evaluating improper heat treatment of

- the (i) waste package outer corrosion barrier,
- the (iii) waste package outer corrosion barrier lid, and
- the (vi) drip shield,

as well as for (iv) the WP-LPB-OPERATOR event in the event tree for evaluating low-plasticity burnishing treatment of the waste package outer corrosion barrier lid and (vii) the DS_EMP_ANN event in the event tree for evaluating improper emplacement of the drip shield.

As noted in Benhardt et al. (1994, Section 3.3), the choice of high, nominal, or low human error probabilities is determined by subjective assessments of the significant influences of human actions, such as the quality of administrative controls, procedures, training, organizational culture, situational stress, and the design of the man–machine interface. This is consistent with Swain and Guttmann's (1983, Chapter 3, Table 3-2) discussion of performance shaping factors (PSFs). They note that situational characteristics (e.g., operating environment and procedures), task and equipment characteristics, psychological and physiological stressors, and individual operator characteristics can influence (positively or negatively) the reliability of human performance. For waste package and drip shield fabrication and handling processes, the use of strict QA requirements derived from 10 CFR 63.142 (SAR Section 1.9) and stringent controls (SAR Table 1.9-9, drip shield and waste package design control parameters) will result in the use of excellent procedures and training and well designed and maintained facilities. Also, as discussed in SAR Section 5.3, industrially accepted codes and standards and regulatory guidance documents related to the training of personnel will be used in repository operations and waste package and drip shield fabrication. As discussed in SAR Section 5.6, plans and procedures for operations, maintenance, surveillance, and periodic testing of structures, systems, or components and processes used in waste package handling, including procedural safety controls, will be written, tested, and approved prior to receipt of waste. In addition, stringent quality assurance requirements will be imposed upon waste package and drip shield fabricators through the Project's quality assurance program. These requirements will be similar to standard nuclear industry practices for safety-related items and equipment. This is demonstrated as discussed in SAR Sections 1.5.2 and 1.3.4 that the waste package and drip shields will be fabricated in accordance with *2001 ASME Boiler and Pressure Vessel Code* (ASME 2001, Section III, Division 1, Subsection NC). Thus, it is reasonable to assume excellent procedures, training, experience, and environment are used during waste package and drip shield fabrication and handling processes. Given that about 12,000 waste packages (SAR Section 1.3.2.4.3.1) and about 11,500 drip shields (SAR Table 1.3.6-1) will be fabricated and emplaced, it is reasonable to use human error probabilities for "Failure of an administrative control" corresponding to "Routine, repetitive circumstances." It is reasonable to expect that few competing signals will be present during these operations. Therefore, it is reasonable to use human error probabilities for "Failure to respond to a compelling signal" corresponding to "Few competing signals" as specified in Benhardt et al. (1994, Table 4).

Although DOE considers the use of the low mean and EF values for the basic events in question to be appropriate, a sensitivity study was run using the nominal mean and EF values for these events (i.e., for the "Failure of administrative control" event, a mean of 5.0×10^{-3} with an EF of 10, and for the "Failure to respond to compelling signal" event, a mean of 1.0×10^{-2} with an EF of 5). The results of this sensitivity are shown in Table 1. As shown in Table 1, although the mean number of early failed waste packages increases by a factor of about three and the number of early failed drip shields increases by a factor of about 10, use of the nominal mean and error factors has a negligible impact on compliance with the individual or groundwater protection standards. This is not unexpected because, as discussed in the response to RAI 3.2.2.1.2.2-001, from a risk-informed perspective, the waste package and drip shield early failure scenarios are minor contributors to the calculated mean annual dose.

Table 1. Results of TSPA-LA Base Case and Sensitivity Analysis Using Nominal Mean and Error Factor Values Compared to Regulatory Limits

	TSPA-LA Base Case	Sensitivity	Limit
Mean number of waste package early failures	1.09	3.52	
Mean number of drip shield early failures	0.018	0.16	
Individual Protection Standard			
Maximum total mean annual dose (in 10,000 years) (mrem)	0.24	0.25	15
Groundwater Protection Standard			
Maximum mean Ra concentration (in 10,000 years) (pCi/L)	1.3×10^{-7}	1.5×10^{-7}	5
Maximum mean Alpha emitter concentration (in 10,000 years) (pCi/L)	6.7×10^{-5}	2.4×10^{-4}	15
Maximum mean annual whole body dose from beta and photon emitters (in 10,000 years) (mrem)	0.059	0.064	4
Maximum mean annual thyroid dose from beta and photon emitters (in 10,000 years) (mrem)	0.26	0.29	4

In addition, as shown in SAR Table 2.2-8, the dominant contributor (by about two orders of magnitude) to the probability of criticality is from seismic activity. As discussed in SAR Section 2.2.1.4.1, the early waste package failures are considered in the nominal case for criticality screening. Therefore, because the increase in the mean number of early failures in the sensitivity study is considerably less than two orders of magnitude, there is no significant impact on the probability of criticality.

In conclusion, the use of the low mean value and error factor (EF) values from Benhardt et al. (1994, Table 4) is appropriate and justified.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

10 CFR 63. 2008. Energy: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada. Internet Accessible.

ASME (American Society of Mechanical Engineers) 2001. *2001 ASME Boiler and Pressure Vessel Code*. New York, New York: American Society of Mechanical Engineers. TIC: 251425.

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.

BSC (Bechtel SAIC Company) 2007. *Yucca Mountain Project Engineering Specification Prototype Drip Shield*. 000-3SS-SSE0-00100-000 REV 000. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071206.0013.

BSC 2008. *Q-List*. 000-30R-MGR0-00500-000-004. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080312.0037.

SNL (Sandia National Laboratories) 2007. *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure*. ANL-EBS-MD-000076 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070629.0002; DOC.20071003.0015; LLR.20080311.0094; DOC.20080918.0002.

Swain, A.D. and Guttmann, H.E. 1983. *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report*. NUREG/CR-1278. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 246563.

RAI: Volume 3, Chapter 2.2.1.2.2, Number 5

Provide additional justification for the probability value for a camera not detecting damage to an emplaced waste package.

Basis: DOE proposes to use a camera to remotely detect damage to the waste package surface that may occur during emplacement (SNL, 2007a, as referenced in SAR, Section 2.3.6.6.3.2.6). For the probability that the damage is not detected, DOE gives a mean value of 2.50×10^{-3} and error factor of 3. This value is defined as a human error event with a value taken from Swain and Guttmann (1983): the "error of commission in check-reading analog meter." DOE has not sufficiently justified how the error probability for "check-reading analog meter" appropriately estimates the error in using a camera to examine the waste package surface for damage. In addition, the error estimate has not considered the potential for mechanical failure of the camera or such factors as a partially degraded visual image. The error estimate and associated uncertainty also does not account for the potential inability of the camera to visualize the entire waste package surface, including the part resting on the pallet.

References:

SNL (2007a) Analysis of Mechanisms for Early Waste Package/Drip Shield Failure. ANL-EBS-MD-000076 Rev00. Las Vegas, Nevada: Sandia National Laboratories

Swain, A.D. and Guttmann, H.E. 1983. Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report. NUREG/CR-1278. Washington, D.C.: U.S.

1. RESPONSE

This response first discusses inspections that take place during emplacement to justify the selected probability for a camera not detecting damage to a waste package during emplacement. Next, the response discusses the potential for mechanical failure of the camera and how this potential could result in an additional event sequence leading to undetected defects in the waste package outer corrosion barrier, followed by an assessment of the effect on the estimated probability for waste package early failure if mechanical failure of sensors had been included as a separate event sequence. Finally, NRC's question about visualization of the entire waste package surface is addressed in the response to RAI 3.2.2.1.2.2-006. The assessment presented in this response demonstrates that inclusion of the possibility of camera faults has a negligible effect on the analysis of early failures reported in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007), and the use of a mean value of 2.50×10^{-3} and error factor of 3 is justified.

1.1 INSPECTIONS DURING EMPLACEMENT

DOE's response to RAI 3.2.2.1.2.2-006 summarizes the inspections that could detect damage to waste packages during emplacement of the waste packages in the repository. Empty waste packages are inspected for damage prior to placement in the waste package transfer trolley (SAR Section 1.2.1.3). During transfer of the sealed waste package from the waste package transfer trolley to the transport and emplacement vehicle, waste packages are again inspected. A camera will be used to inspect sealed waste packages for unacceptable surface marring (SAR Section 1.2.4.2.4.2). Cameras will be of sufficiently high resolution to allow operators to properly determine the acceptability of the waste package outer corrosion barrier prior to transport and emplacement in the repository.

Given that the camera operates correctly, interpretation of the display resulting from a camera inspection of the waste package outer corrosion barrier can be modeled by an operator monitoring a visual display on which the view of an unmarred portion of a waste package outer barrier is the normal condition. Failure of this inspection occurs when the operator fails to note a deviant condition on the display (i.e., view of a marred portion of the waste package outer corrosion barrier); thus, classifying this error as an error of commission is appropriate. Moreover, monitoring the analog visual display for deviant conditions is similar to monitoring an analog meter for a reading within acceptable limits, because no quantitative information is recorded, and because the operator will examine more closely any apparent deviation from the normal condition (i.e., view of an unmarred surface). Therefore, "check-reading of an analog meter" is an appropriate model for the probability of error in the inspection process during emplacement, and this probability is characterized by the distribution given in Swain and Guttmann (1983, Table 20-11, Item 3) (i.e., check-reading of an analog meter with difficult-to-see limit marks). Because the handling of the waste package outer corrosion barrier will be accomplished under stringent controls (SAR Table 1.9-9, waste package design control parameters), waste packages will be individually inspected at a pace that is comfortable for the operator and without competing demands for the operator's attention during the inspection. Thus, the level of stress on the operator during the inspection is expected to be low, and no performance-shaping factors are applied to the selected distribution.

Faults in the camera (mechanical or other) could affect the rate of error in the inspection of waste package outer surfaces during emplacement. It is reasonable to expect an operator to detect and correct any fault that would prevent the camera from imaging the waste package outer barrier or which result in an obviously incorrect image. For the purpose of this response, the probability of a camera fault is conservatively selected to be 0.01 per waste package inspection; current technology produces complex electro-optic systems including high-resolution cameras with mean time between failures (MTBF) for the system exceeding thousands of operating hours (Jane's Electro-Optic Systems, available online at http://jeos.janes.com/public/jeos/index.shtml).

As part of the operating procedure for the waste package inspection, the operator would check the camera display against a known image or calibration target to verify correct camera function. Thus, the camera faults that may affect inspection of the waste package outer barrier are those which result in display of an incorrect image which the operator fails to notice is incorrect. Because the handling of the waste package outer corrosion barrier will be accomplished under stringent controls (SAR Table 1.9-9, waste package design control parameters), it is reasonable to expect that an operator who notes an incorrect image on the camera display would take steps to address the fault and to obtain a correct image. Therefore, operator failure to recognize a camera fault can be modeled as "check-reading of an analog display" without any performance-shaping factors.

In summary, the occurrence of undetected faults in the camera is quantified by scaling the distribution given in Swain and Guttmann (1983, Table 20-11, Item 3) (i.e., check-reading of an analog meter with difficult-to-see limit marks) by the probability of a camera fault occurring during a waste package inspection (0.01). The possibility of faults in the camera presents an event sequence that may result in undetected flaws in waste package outer corrosion barrier that was not included in the analysis reported in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007). The effect of including the potential for faults in the camera on the estimated probability of waste package early failure is discussed in Section 1.2 of this response.

1.2 EFFECT OF INCLUDING SENSOR FAULTS ON EARLY FAILURE PROBABILITY

Figure B-14 in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007) presents the event tree used to compute the contribution to the probability of waste package early failure from the occurrence of damage during waste package emplacement, and the failure to detect such damage during inspections. This event tree does not include an event sequence that accounts for the possibility that faults in the camera may result in undetected flaws in the waste package outer corrosion barrier. Figure 1 below illustrates an alternate event tree that includes an event sequence accounting for camera faults.

Evaluation of the event tree shown in Figure 1 yields a mean probability of occurrence of damage to the waste package outer corrosion barrier from mishandling, the failure to detect such damage during inspections, of 9.70×10^{-7}, equivalent to the mean probability of 9.71×10^{-7} estimated by the Monte Carlo analysis performed with SAPHIRE (Table 1 in DOE's response to RAI 3.2.2.1.2.2-002) that results from the event tree presented in Figure B-14 of *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007). Waste package mishandling is one of six possible event sequences that contribute to the overall probability of waste package early failure. If the increase in the probability of damage from waste package mishandling due to including the possibility for faults in the camera was included, the sum of the mean probabilities over all event sequences would not change from 1.13×10^{-4} (SAR Section 2.3.6.6.3.2.7) Thus, inclusion of the possibility of camera faults has a negligible effect on the analysis of early failures reported in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007).

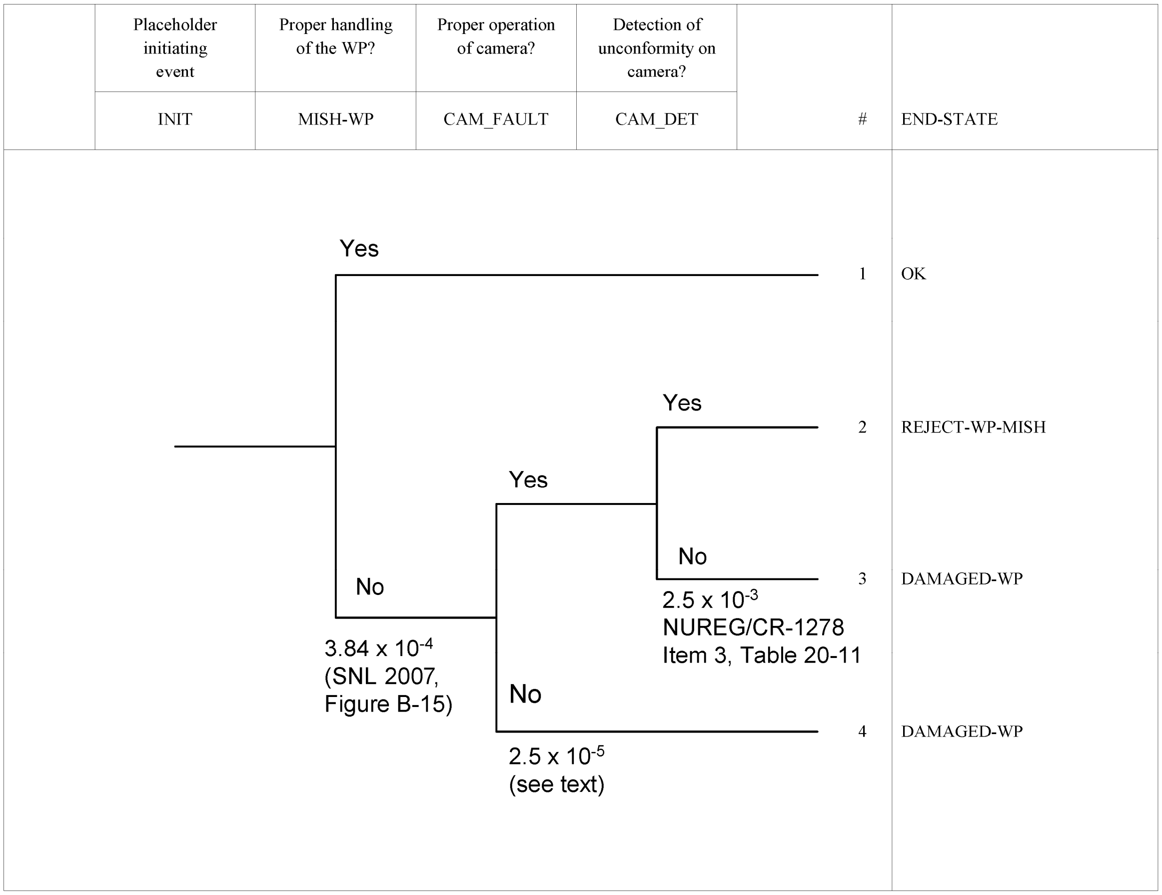

Figure 1. Alternate Event Tree for Evaluating Mishandling of the Waste Package Outer Corrosion Barrier

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

SNL (Sandia National Laboratories) 2007. *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure.* ANL-EBS-MD-000076 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070629.0002; DOC.20071003.0015; LLR.20080311.0094; DOC.20080918.0002.

Swain, A.D. and Guttmann, H.E. 1983. *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report.* NUREG/CR-1278. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 246563.

RAI: Volume 3, Chapter 2.2.1.2.2, Number 7

Provide additional justification for using nuclear fuel assembly handling as an analogue for estimating the probability of waste package damage during handling and emplacement.

Basis: To evaluate the probability of damage during waste package emplacement, DOE used damage to nuclear fuel assemblies during their handling as an analogue because "…fuel assembly-handling activities are performed in a nuclear environment representative of the highly controlled conditions under which handling of the waste package is expected to occur" (SNL 2007a, as referenced in SAR Section 2.3.6.6.3.2.6). However, waste package handling during emplacement will involve numerous above-ground and underground operations that are not encountered during fuel assembly handling. DOE has not provided an appropriate technical basis to conclude that: (i) the tasks, in terms of type and complexity, associated with waste package emplacement are analogous to fuel assembly handling and (ii) the operating environment in the repository during emplacement is comparable to that in which fuel assemblies are handled in a nuclear power plant.

Reference:

SNL (Sandia National Laboratories). 2007a. Analysis of Mechanisms for Early Waste Package/Drip Shield Failure. ANL-EBS-MD-000076 REV 00. Las Vegas, Nevada: Sandia National Laboratories.

1. RESPONSE

While waste package handling will involve several above-ground and underground operations that are not encountered during nuclear fuel assembly handling, the repository environment and the design controls used are similar to those that would be used in a nuclear plant. Any repository operation involving waste package handling is subject to strict QA control requirements derived from 10 CFR Part 63.142 (SAR Section 1.9). In addition, nuclear fuel assembly handling is a reasonable proxy for estimating the probability of waste package damage during handling and emplacement. Both nuclear fuel assembly and waste package handling processes involve lifting and transport of heavy loads in a highly controlled nuclear environment. Fuel assemblies must be lifted from confined spaces (e.g., transportation casks, the storage pool) and lowered into confined spaces (e.g., transportation casks, the reactor core) while minimizing contact with other objects or exceeding inertial limits. These tasks, in terms of type and complexity, are similar to loading of the waste packages and transporting of waste packages during emplacement. Although a subsurface emplacement drift differs from a reactor plant, the handling operations for the waste package are similar to fuel assembly handling because both require mechanical handling and remote operations. The Project will use the guidance contained in industrially accepted codes and standards and regulatory documents (e.g., NUREG-0612 (NRC 1980) and ANSI N14.6-1993) (SAR Table 1.2.2-12; SAR Table 1.3.2-5) for the handling of heavy loads such as waste packages. These same industrially accepted codes and standards and regulatory guidance documents are also used in handling heavy loads, such as a spent nuclear fuel assembly and its associated handling tool, in a nuclear plant (NRC 1980, p. 1-2). As discussed in SAR Section 5.3, industrially accepted codes and standards and regulatory guidance documents (e.g., ANSI/ANS-3.1-1993 (1999), ANSI/ANS-3.4-1996, ASTM E 1168-95) related to the training of personnel will be used in repository operations. As discussed in SAR Section 5.6, plans and procedures for operations, maintenance, surveillance, and periodic testing of structures, systems, or components and processes used in waste package handling, including procedural safety controls, will be written, tested, and approved prior to receipt of waste. The use of industrially accepted codes and standards and regulatory guidance documents as well as written plans and procedures will result in the repository operating environment being similar to that in which heavy loads are handled in a nuclear plant.

A fault tree composed of eight generic basic events (which act as surrogates for operations that could lead to potential waste package damage) was used in *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure* (SNL 2007). Each of the basic events was assigned a mean probability of damage value of 4.8×10^{-5} (based on analysis of the probability of damage to a spent fuel assembly during handling) and an error factor of 10 to provide an uncertainty range (SNL 2007, Section 6.3.6). If at least one of these generic events is triggered, handling damage of the waste package is considered to have occurred. Given that the environment and controls used in the repository are similar to those used in a nuclear plant, the mean probability of damage during waste package handling is expected to be on the same order of magnitude as the probability of damage to a spent fuel assembly during handling. Details of the waste package handling process are summarized in RAI response 3.2.2.1.2.2-006. While the eight generic events used in the fault tree for waste package handling damage were not associated with a particular handling process, they can be considered to represent the possibility of waste package damage from processes such as the waste package being tilted in an upward position, being down-ended, being placed onto the waste package emplacement pallet, or being moved onto or off of the transport and emplacement vehicle, etc., as described in RAI response 3.2.2.1.2.2-006.

SAR Section 2.3.6.6.4.2 demonstrates that the estimated overall waste package early failure probabilities are similar to failure data from boilers, pressure vessels, nuclear fuel rods, underground storage tanks, radioactive cesium capsules, and SNF dry storage casks. As shown in RAI response 3.2.2.1.2.2-002, the probability of waste package handling damage (referred to as handling error in Table 1 of RAI response 3.2.2.1.2.2-002) is a minor contributor to the overall probability of waste package early failure.

Therefore, the Project's treatment of waste package handling damage is appropriate.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

ANSI N14.6-1993. *American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More.* New York, New York: American National Standards Institute. TIC: 236261.

ANSI/ANS-3.1-1993. 1999. *American National Standard for Selection, Qualification, and Training of Personnel for Nuclear Power Plants*. La Grange Park, Illinois: American Nuclear Society. TIC: 235767.

ANSI/ANS-3.4-1996. *American National Standard for Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants*. La Grange Park, Illinois: American Nuclear Society. TIC: 251478.

ASTM E 1168-95. *Standard Guide for Radiological Protection Training for Nuclear Facility Workers*. West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 241268.

NRC (U.S. Nuclear Regulatory Commission) 1980. *Control of Heavy Loads at Nuclear Power Plants*. NUREG-0612. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 209017.

SNL (Sandia National Laboratories) 2007. *Analysis of Mechanisms for Early Waste Package/Drip Shield Failure*. ANL-EBS-MD-000076 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070629.0002; DOC.20071003.0015; LLR.20080311.0094; DOC.20080918.0002.