E6.6.3.4 Quantitative Analysis (Step 7)

Once the HFE scenarios and the unsafe actions within them are scoped out, it is then possible to review them in detail and apply the appropriate quantification methodology in each case that permits an HEP to be calculated for each HFE. Stated another way, each HFE is quantified through the analysis and combination of the contributing HFE scenarios. Dependencies between the unsafe actions and equipment responses within each scenario and across the scenarios are carefully considered in the quantification process.

This section provides a description of the quantitative analysis performed, structured hierarchically by each HFE category (identified by a basic event name); the HFE scenario; and then the unsafe actions under each scenario, as previously documented in Table E6.6-3.

Prior to the scenario-specific quantification descriptions, a listing is provided of the values used in the quantification that are common across many of the HFE scenarios.

In generating the final HEP values, the use of more than a single significant figure is not justified given the extensive use of judgment required for the quantification of the individual unsafe actions within a given HFE. For this reason, all calculated final HEP values are reduced to one significant figure. When doing this, the value is always rounded upwards to the next highest single significant figure.

E6.6.3.4.1 Common Values Used in the HFE Detailed Quantification

There are some mechanical failures that combine with unsafe actions to form HFEs. In general, these mechanical failures are independent of the specific HFE scenario, and so they can be quantified independently. These values are presented in this section.

Interlock Failures—There are a number of interlock failures in the HFE scenarios. While the status of these events can affect subsequent events in the scenarios in different ways, the likelihood of this event occurring is independent of the scenario. This event is an equipment failure and does not have a human component to its failure rate. The demand failure rate for an interlock, from Attachment C, Table C4-1, is approximately 2.7E−05 per demand.

Interlock fails to perform function = 2.7E−05

Load Cell Failure—This mechanical failure was used for the load sensor used to evaluate if there is a loaded WPTT in the positioning room. The mechanical failure probability for a load cell, represented by a pressure sensor from Attachment C, Table C4-1, is approximately 4.0E−03/demand.

Load cell fails = 4.0E−03

E6.6.3.4.2 Quantification of HFE Scenarios for 51A-OpDirExpose3-HFI-NOD: Operator Causes Direct Exposure during TEV Loading

E6.6.3.4.2.1 HFE Group #6 Scenario 1(a) for 51A-OpDirExpose3-HFI-NOD

1. A crew member remains on the second floor of the Waste Package Loadout Room after an evacuation is ordered OR a WPTT operator fails to order an evacuation.

2. Radiation Protection Worker fails to check if the room is empty OR fails to recognize that someone is still in the room.

3. A crew member fails to notice that loadout is occurring OR a crew member fails to exit the room in time to avoid exposure.

A Crew Member Remains on the Second Floor of the Waste Package Loadout Room after an Evacuation Is Ordered—Prior to moving a loaded waste package to the Waste Package Loadout Room, there may be maintenance or (empty) waste package preparation activities going on in the Waste Package Loadout Room. Once the WPTT is ready to be moved to the Waste Package Loadout Room, the WPTT operator makes an announcement for all personnel to leave the Waste Package Loadout Room.

Even if the WPTT operator notifies workers to leave the Load out Room, it is possible that they would not do so. The action of leaving is quite simple, but the communication of the request could be missed by virtue of the person being engrossed in a task. This can be represented by NARA GTT D1, adjusted for the following EPCs:

- GTT D1: Verbal communication of safety-critical data. The baseline HEP is 0.006.

- EPC 4: Low signal-to-noise ratio. This usually applies to a proliferation of information, but it can be applied to masking by any distracting mechanism. The full effect is ×10, which applies to significant levels of distraction. In this case, the level of distraction would be small as there would not be a significant amount of machine noise to mask the facility paging system announcement. The APOA is judged to be minimal, and thus is set to 0.1.

Using the HEP equation yields:

$$\text{Crew member remains on the second floor of the Waste Package Loadout Room after an evacuation is ordered} = 0.006 \times [(10-1) \times 0.1 + 1] = 0.01 \qquad \text{(Eq. E-23)}$$

WPTT Operator Fails to Order Evacuation—The WPTT operator is required to announce that all personnel are to leave the Waste Package Loadout Room prior to initiating movement of the WPTT into the room. This action is part of the process procedure. The operator is required to make the announcement whether or not it is believed that anyone is in the room, so this can be represented by CREAM (Ref. E8.1.18) execution CFF E5, adjusted for the following CPCs with value not equal to 1.0:

- CFF E5: Action missed, not performed (omission). The baseline HEP is 0.03.

- CPC "Working Conditions": The working conditions for the operator are in the IHF Control Room with a favorable environment. The CPC for advantageous working conditions for an execution task is 0.8.

- CPC "Availability of Procedures": With regard to the notification step, the procedures and checklist clearly list that this task needs to be performed. The CPC for appropriate availability of procedures for an execution task is 0.8.

- CPC "Available Time": There is more than enough time to successfully perform this task. The CPC for adequate available time for an execution task is 0.5.

- CPC "Adequacy of Training/Preparation": This is a routine task that is clearly trained and emphasized in training. Because it is routine, there is a high level of experience. The CPC for adequate training and high experience for an execution task is 0.8.

Applying these factors yields the following:

$$\text{WPTT operator fails to order an evacuation} = 0.03 \times 0.8 \times 0.8 \times 0.5 \times 0.8 = 0.008$$

Radiation Protection Worker Fails to Check if Room Is Empty—This is an EOO by the radiation protection worker (or another pre-designated person) who fails to personally ensure that the Waste Package Loadout Room has been cleared of personnel. This is considered to be represented by CREAM (Ref. E8.1.18) execution CFF E5, adjusted for the following CPCs:

- CFF E5: Action missed, not performed (omission). The baseline HEP is 0.03.

- CPC "Working Conditions": The working conditions are in the Waste Package Loadout Room with a less favorable environment than the IHF Control Room due to controlled access. The CPC for incompatible working conditions for an execution task is 2.0.

- CPC "Availability of Procedures": With regard to the notification step, the procedures and checklist clearly list that this task needs to be performed. The CPC for appropriate availability of procedures for an execution task is 0.8.

- CPC "Available Time": It is anticipated that there is some time pressure to successfully perform this task so that loadout can proceed. The CPC for temporarily inadequate available time for an execution task is 1.0.

- CPC "Adequacy of Training/Preparation": This is a routine task that is clearly trained and emphasized in training. Because it is routine, there is a high level of experience. The CPC for adequate training and high experience for an execution task is 0.8.

Applying these factors yields the following:

$$\text{Radiation protection workers fail to check if the room is empty} = 0.03 \times 2.0 \times 0.8 \times 1.0 \times 0.8 = 0.04$$

Radiation Protection Worker Fails to Recognize that Someone Is Still in the Room—This EOC is considered to occur due to a lack of attention or perhaps a distraction that causes the radiation protection worker to fail to perform the check properly. It is considered to be covered by HEART (Ref. E8.1.28) Generic Task (D), modified by the following EPC:

- GT (D): Fairly simple task performed rapidly or given scant attention with a baseline HEP of 0.09.

- EPC 17: Inadequate checking (HEART EPC 17). Little or no independent checking. The full effect is ×3. In this case, since the radiation protection worker is completely unsupervised, the APOA is judged to be complete and is set at 1.0.

$$\text{Radiation protection worker fails to recognize that someone is still in the room} = 0.09 \times [(3-1) \times 1.0 + 1]) = 0.27 \quad \text{(Eq. E-24)}$$

Crew Member Fails to Notice that Loadout Is Occurring—If someone is left in the Waste Package Loadout Room once the WPTT operator begins tilt-down, the crew member in the Waste Package Loadout Room has a few minutes before actually getting a direct exposure (after the waste package has been tilted down and the transfer carriage removed). All exits from the Waste Package Loadout Room are clearly marked and unlocked from the inside. All crew involved in TEV loading operations are trained and aware of the severe consequences associated with exposure to a bare waste package.

This error most closely corresponds to the CREAM (Ref. E8.1.18) observation error O3 ("Observation Not Made").

- CFF O3: Observation not made (omission). The baseline HEP is 0.001.

- CPC "Working Conditions": The working conditions are in the Waste Package Loadout Room with a less favorable environment than the IHF Control Room due to controlled access. The CPC for incompatible working conditions for an observation task is 2.0.

- CPC "Available Time": It is anticipated that there is some time pressure to successfully perform this task so that load out can proceed. The CPC for temporarily inadequate available time for an observation task is 1.0.

- CPC "Adequacy of Training/Preparation": This is a task that is clearly trained, and the risks of loadout are emphasized in training. Because it is performed frequently, there is a high level of experience. The CPC for adequate training and high experience for an observation task is 0.8.

Applying these factors yields the following:

$$\text{Crew member fails to notice that loadout is occurring} = 0.001 \times 2.0 \times 1.0 \times 0.8 = 0.002$$

Crew Member Fails to Exit the Room in Time to Avoid Exposure—Training indicates the need for action in such a case. A few minutes are available to exit through clearly marked doors. This can be represented by NARA GTT C1, adjusted for the following EPCs:

- GTT C1: Simple response to a range of alarms/indications providing a clear indication of the situation (simple diagnosis required). Response might be a direct execution of simple actions or initiating other actions separately assessed. The baseline HEP is 0.0004.

- EPC 3: Time pressure. The full effect is ×11, which corresponds to the operator having just enough time. In this case, there are some minutes left to act. The APOA anchor for 0.5 is that the operator must work at a fast pace. The situation is judged to be between these two anchors, and thus the APOA is set to 0.75.

Using the NARA HEP equation yields the following:

$$\text{Crew member fails to exit the room in time to avoid exposure} = 0.0004 \times [(11-1) \times 0.75 + 1] = 0.003 \quad \text{(Eq. E-25)}$$

HEP Calculation for Scenario 1(a)—The events in the HEP model for Scenario 1(a) are presented in Table E6.6-4.

Table E6.6-4. HEP Model for HFE Group #6 Scenario 1(a) for 51A-OpDirExpose3-HFI-NOD

Designator	Description	Probability
A	A crew member remains on the second floor of the Waste Package Loadout Room after an evacuation is ordered	0.01
B	A WPTT operator fails to order an evacuation	0.008
C	Radiation Protection Worker fails to check if the room is empty	0.04
D	Radiation Protection Worker fails to recognize that someone is still in the room	0.27
E	A crew member fails to notice that loadout is occurring	0.002
F	A crew member fails to exit the room in time to avoid exposure	0.003

NOTE: WPTT = waste package transfer trolley.

Source: Original

The Boolean expression for this scenario follows:

$$(A + B) \times (C + D) \times (E + F) = (0.01 + 0.008) \times (0.04 + 0.27) \times (0.002 + 0.003) = (0.018) \times (0.31) \times (0.005) = 3E-5 \quad \text{(Eq. E-26)}$$

E6.6.3.4.2.2 HFE Group #6 Scenario 1(b) for 51A-OpDirExpose3-HFI-NOD

1. A crew member requests reentry into the Waste Package Loadout Room.
2. The supervisor agrees to allow access.

Crew Member Requests Re-entry into the Waste Package Loadout Room—TEV loading is a normal part of operations that takes less than a few hours and is performed about twice a week.

Workers are trained not to enter the Waste Package Loadout Room unless necessary for a prescheduled activity. There is a posted schedule that all the workers, including the supervisor, are aware of. Also, during loadout, there are indicators in the IHF Control Room and outside both the personnel access door and the shield door that turn on when TEV loading is in progress. These indicators, however, are non-ITS equipment, and no credit is given for their function.

Entry would have to be for some perceived urgent need in order to countermand training. This is believed to be best represented by NARA GTT A4, adjusted for the following EPCs:

- GTT A4: Judgment needed for appropriate procedure to be followed, based on interpretation of a situation that is covered by training at appropriate intervals. The baseline HEP is 0.006.

- EPC 14: Conflict between immediate and long-term objectives. The full effect is ×2.5, which corresponds to deciding between two extremely significant problems, one of which is clearly more urgent than the other. The APOA anchor for 0.5 can be interpreted as suggesting situations of balancing an urgent plant need with an urgent personal need. The APOA anchor for 0.1 is that an individual has immediate personal needs, but there is an obvious safety task that requires completion. Other reasons for reducing from full effect include training and procedures that dictate a hierarchy of goals (which, in this case, may or may not actually help). The situation in this case is judged to be between 0.1 and 0.5, and thus the APOA is set to 0.3.

Using the NARA HEP equation yields the following:

$$\text{Crew member requests reentry into the Waste Package Loadout Room} = 0.006 \times [(2.5-1) \times 0.3 + 1] = 0.009 \qquad \text{(Eq. E-27)}$$

It should be noted that there are other possible unsafe actions for this scenario, such as the supervisor either failing to request a stop of the loadout operation, failing to verify that the operation has stopped before opening the door, or prematurely restarting the operation while the worker is still in the Waste Package Loadout Room. In addition, it is possible that the WPTT operator could fail to stop the loadout operation as requested. However, these actions were considered much less likely than the primary unsafe actions cited above and, further, would only serve to drive down the overall scenario HEP value. Therefore, the analysts decided to conservatively constrain the analysis to the unsafe actions quantified above.

HEP Calculation for Scenario 1(b)—The events in the HEP model for Scenario 1(b) are presented in Table E6.6-5.

Table E6.6-5. HEP Model for HFE Group #6 Scenario 1(b) for 51A-OpDirExpose3-HFI-NOD

Designator	Description	Probability
A	Crew member requests reentry into the Waste Package Loadout Room	0.009
B	Supervisor agrees to allow access	0.0002

Source: Original

The Boolean expression for this scenario follows:

$$A \times B = 0.009 \times 0.0002 = 2E-06 \qquad \text{(Eq. E-28)}$$

E6.6.3.4.2.3 HFE Group #6 Scenario 1(c) for 51A-OpDirExpose3-HFI-NOD

1. Personnel access shield door is left open.
2. Interlock or load sensor fails and a WPTT enters the Waste Package Loadout Room.

Personnel Access Door Left Open—Before the WPTT is moved from the Waste Package Positioning Room to the Waste Package Loadout Room, a designated radiation protection worker checks that the area is free from personnel and ensures that all the personnel access doors are closed and locked. (Note: If the personnel access doors are left unlocked, a person could potentially enter the room and incur direct exposure, but this is bounded by the current scenario.)

In this scenario, the unsafe action that occurs is that one of the shield doors is left open (i.e., the shield is not fully in place). This scenario also includes the radiation protection worker being successful in ensuring that all personnel have left the room. (The unsafe action of failing to ensure that the room is empty is addressed in Scenario 1(a).)

The failure to recognize that an access door is left open is believed to be best represented by NARA GTT B3: Set system status as part of routine operations using strict administratively controlled procedures. The baseline HEP is 0.0007.

In addition, the WPTT operator would have to fail to verify by camera that the door was left open and fail to check the tag in/tag out board to address accountability for the location of personnel. This error is presumed to have medium dependency. Based on Table 20-21 of THERP (Ref. E8.1.26), for a baseline HEP of <0.01 and medium dependence, the appropriate dependence value would be Item (3)(a) or 0.15.

$$\text{Personnel access shield door left open} = 0.0007 \times 0.15 = 0.0001$$

Interlock or Load Cell Fails and WPTT Enters the Waste Package Loadout Room—There is an interlock between the personnel access shield doors and the Waste Package Positioning Room shield door. If there is a loaded WPTT in the Waste Package Positioning Room (load sensor), in order for the Waste Package Positioning Room shield door to open (to allow the WPTT to move into the Waste Package Loadout Room), the personnel access doors must be closed and locked. A direct exposure would occur if the interlock were to fail since nothing would stop the WPTT operator from moving the WPTT into the Waste Package Loadout Room, and nothing would prevent radiation from escaping the room.

The mechanical failure probability for an interlock, from Attachment C, Table C4-1, is approximately 2.7E−5/demand.

$$\text{Interlock fails} = 2.7E-5$$

The mechanical failure probability for a load cell, represented by a pressure sensor from Attachment C, Table C4-1, is approximately 4.0E−03/demand.

Load cell fails = 4.0E−03

Interlock or load cell fails = 2.7E−05 + 4.0E−03 = 0.004

HEP Calculation for Scenario 1(c)—The events in the HEP model for Scenario 1(c) are presented in Table E6.6-6.

Table E6.6-6. HEP Model for HFE Group #6 Scenario 1(c) for 51A-OpDirExpose3-HFI-NOD

Designator	Description	Probability
A	Personnel access shield door is left open	0.0001
B	Interlock or load cell fails, and a WPTT enters the Waste Package Loadout Room	4E−03

NOTE: WPTT = waste package transfer trolley.

Source: Original

The Boolean expression for this scenario follows:

$$A \times B = 0.0001 \times 4E{-}03 = 4E{-}07 \qquad \text{(Eq. E-29)}$$

HEP for HFE 51A-OpDirExpose3-HFI-NOD—The Boolean expression for the overall HFE (all scenarios) follows:

$$\text{51A-OPDIREXPOSE3-HFI-NOD} = \text{HEP 1(a)} + \text{HEP 1(b)} + \text{HEP 1(c)} = 3E{-}05 + 2E{-}06 + 4E{-}07 = 3E{-}05 \qquad \text{(Eq. E-30)}$$

E6.6.4 Results of Detailed HRA for HFE Group #6

The final HEPs for the HFE that required detailed analysis in HFE Group #6 is presented in Table E6.6-7 (with the original preliminary value shown in parentheses).

Table E6.6-7. Summary of HFE Detailed Analysis for HFE Group #6

HFE	Description	Final Probability
51A-OpDirExpose3-HFI-NOD	Operator causes direct exposure while loading TEV	3E−05 (1E−3)

NOTE: TEV = transport and emplacement vehicle.

Source: Original

E7 RESULTS: HUMAN RELIABILITY ANALYSIS DATABASE

Table E7-1 presents a summary of all of the human failures identified in this analysis, and provides a link between the HFE group and the ESD in which the human failure is modeled.

Table E7-1. HFE Data Summary

Basic Event Name	HFE Description	ESD	HFE Group	Basic Event Mean Probability	Error Factor	Type of Analysis
51A-Liddisplace1-HFI-NOD	Operator inadvertently displaces cask lid during preparation activities	12	3	N/A[b]	N/A	Omitted from Analysis
51A-OpCaskDrop01-HFI-NOD	Operator drops cask during cask preparation activities	N/A	3	N/A[b]	N/A	Omitted from Analysis
51A-OpClCTMGate1-HFI-NOD	Operator inappropriately closes slide or port gate during vertical canister movement and continues lifting	7	4	1.00E−03	5	Preliminary
51A-OpCollide001-HFI-NOD	Operator causes low-speed collision of auxiliary vehicle with RC, TT, or CTT	1, 2, 3, 4	2, 3	3.00E−03	5	Preliminary
51A-OpCraneIntfr-HFI-NOD	Operator causes WP handling crane to interfere with TEV or WPTT	11	6	1.00E−04	10	Preliminary
51A-OpCTCollide2-HFI-NOD	Operator causes low-speed collision of CTT during transfer from preparation station to Unloading Room	5	3	1.00E−03	5	Preliminary
51A-OpCTMDrInt01-HFI-COD	Operator lifts object or canister too high with CTM (two-block)	7	4	1.0	N/A	Preliminary
51A-OpCTMdrop001-HFI-COD	Operator drops object onto canister during CTM operations	7	4	4.00E−07	10	Detailed
51A-OpCTMdrop002-HFI-COD	Operator drops canister during CTM operations	7	4	2.00E−04	10	Detailed
51A-OpCTMImpact1-HFI-COD	Operator moves the CTM while canister or object is below or between levels	7	4	4.00E−08	10	Detailed
51A-OpCTMImpact2-HFI-COD	Operator causes canister impact with lid during CTM operations (HLW)	7	4	N/A[b]	N/A	Omitted from Analysis
51A-OpCTMImpact5-HFI-COD	Operator causes canister Impact with SSC during CTM operations (all)	7	4	1.0	N/A	Preliminary
51A-OpCTTImpact1-HFI-NOD	Operator causes an impact between cask and SSC due to crane operations	1,2, 3, 4	2, 3	3.00E−03	5	Preliminary

Table E7-1. HFE Data Summary (Continued)

Basic Event Name	HFE Description	ESD	HFE Group	Basic Event Mean Probability	Error Factor	Type of Analysis
51A-OpDirExpose1-HFI-NOD	Operator causes direct exposure during CTM activities (all waste forms)	12	4	1.0	N/A	Preliminary
51A-OpDirExpose2-HFI-NOD	Operator causes direct exposure during CTM activities (transfer into a WP)	12	4	1.00E−04	10	Preliminary
51A-OpDirExpose3-HFI-NOD	Operator causes direct exposure during TEV loading	12	6	3.00E−05	10	Detailed
51A-OpFailRstInt-HFI-NOM	Operator fails to restore interlock after maintenance	12	4, 6	1.00E−02	3	Preliminary
51A-OpFailSG-HFI-NOD	Operator fails to close the CTM slide gate before lifting shield skirt (while the canister is inside the bell; direct exposure)	12	4	1.00E−3	5	Preliminary
51A-OpFLCollide1-HFI-NOD	Operator causes high-speed collision of auxiliary vehicle with RC, TT, or CTT	1, 2, 3, 4	2, 3	1.0	N/A	Preliminary
51A-OpImpact0000-HFI-NOD	Operator causes impact of cask during transfer from preparation station to Unloading room	5	3	N/A[b]	N/A	Omitted from Analysis
51A-OpNoDiscoAir-HFI-NOD	Operator fails to disconnect air supply from CTT in the Unloading Room	7	4	1.00E−03	5	Preliminary
51A-OpNoUnBolt00-HFI-NOD	Operator fails to fully unbolt the cask lid before moving CTT into the Unloading Room (HLW)	7	4	1.00E−03	5	Preliminary
51A-OpNoUnBoltDP-HFI-NOD	Operator fails to fully unbolt the cask lid before moving CTT into the Unloading Room (Naval Cask)	7	4	N/A[b]	N/A	Omitted from Analysis
51A-OpNVYShield1-HFI-COW	Operator inappropriately removes naval shield ring (direct exposure)	12	3	3.00E−04	5	Preliminary
51A-OpRCCollide1-HFI-NOD	Operator causes low-speed collision between RC and facility SSCs	1	1	3.00E−03	5	Preliminary
51A-OpRCIntCol01-HFI-NOD	Operator causes high-speed collision between RC and facility SSCs	1	1	1.0	N/A	Preliminary
51A-OpRCIntCol2-HFI-NOD	Operator causes MAP to collide into RC	1	1	1.0	N/A	Preliminary
51A-OpSDClose001-HFI-NOD	Operator closes shield door on waste form in conveyance	6	OA (1,3,6)	1.0	N/A	Preliminary

Table E7-1. HFE Data Summary (Continued)

Basic Event Name	HFE Description	ESD	HFE Group	Basic Event Mean Probability	Error Factor	Type of Analysis
51A-OpShieldRing-HFI-NOD	Operator fails to install WP shield ring in WPTT (direct exposure)	12	6	1.00E−04	10	Preliminary
51A-OpSpurMove01-HFI-NOD	Operator causes spurious movement of CTT in the Preparation Area	1, 2, 3, 4	2, 3	1.00E−04	10	Preliminary
51A-OpTEVDrClosd-HFI-NOD	Operator begins WP extraction before TEV doors open	11	6	1.00E−03	5	Preliminary
51A-OpTiltDown01-HFI-NOD	Operator prematurely tilts down the WPTT	7, 8, 10	4, 5, 6	1.0	N/A	Preliminary
51A-OpTipover001-HFI-NOD	Operator causes cask to tip over during cask upending and removal	1, 2	2	1.00E−04	10	Preliminary
51A-OpTipover002-HFI-NOD	Operator causes cask to tip over during cask preparation activities	3, 4	3	1.00E−04	10	Preliminary
51A-OpTTCollide1-HFI-NOD	Operator causes low-speed collision between TT and facility SSCs	1	1	3.00E−03	5	Preliminary
51A-OpTTIntCol01-HFI-NOD	Operator causes high-speed collision between TT and facility SSCs	1	1	1.0	N/A	Preliminary
51A-OpTTIntCol2-HFI-NOD	Operator causes MAP to collide into TT	1	1	1.0	N/A	Preliminary
51A-OpTTRollover-HFI-NOD	Operator causes rollover of TT	1	1	N/A[b]	N/A	Omitted from Analysis
51A-OpWPCollide1-HFI-NOD	Operator causes low-speed collision of WPTT into SSC	8, 10	5, 6	3.00E−03	5	Preliminary
51A-OpWPInnerLid-HFI-NOD	Operator causes direct exposure during WP loading	12	5	1.00E−04	10	Preliminary
51A-OpWPTiltUp01-HFI-NOD	Operator prematurely tilts up the WPTT	11	6	1.0	N/A	Preliminary
51A-OpWPTTSpur01-HFI-NOD	Operator causes spurious movement of WPTT during canister loading	7	4	1.00E−03	5	Preliminary
Crane Drops	Operator drops cask or drops object onto cask during crane operations	1, 2, 3, 4, 9, 11	2, 3, 5, 6	N/A[a]	N/A	Historic Data
Improper WP Closure	Operator damages canister or fails to properly weld the WP	9	5	N/A[b]	N/A	Omitted from Analysis

Table E7-1. HFE Data Summary (Continued)

Basic Event Name	HFE Description	ESD	HFE Group	Basic Event Mean Probability	Error Factor	Type of Analysis
Load too Heavy	Operator causes drop of cask by attempting to lift a load that is too heavy for the crane	N/A	OA	N/A[b]	N/A	Omitted from Analysis
Moderator Introduced into Moderator-Controlled Area	Operator introduces moderator into a moderator-controlled area of the IHF	N/A	OA	N/A[b]	N/A	Omitted from Analysis
RC Derailment	Operator causes the RC to derail	1	1	N/A[a]	N/A	Historic Data
Spurious Movement of CTT during CTM Activities	Operator causes spurious movement of the CTT during CTM activities	7	4	N/A[b]	N/A	Omitted from Analysis
TEV Collision	Operator causes TEV to collide with WP or WPTT	11	6	N/A[b]	N/A	Omitted from Analysis
WPTT Derailment	Operator causes WPTT to derail	8, 10	5, 6	N/A [a]	N/A	Historic Data
WPTT Uncontrolled Tiltdown	Operator causes an uncontrolled tilt down of the WPTT	10	6	N/A[b]	N/A	Omitted from Analysis

NOTE: [a] Historical data was used to produce a probability for this HFE; this is not covered as part of the HRA, but rather addressed in Attachment C.
[b] These HFEs were initially identified, but omitted from analysis for various reasons, including a design change precluding the human failure, or the failure would require a series of unsafe actions in combination with mechanical failures, such that the event is no longer credible. See the appropriate HFE group in Attachment E for a case-by-case justification for these omissions.

CTM = canister transfer machine; CTT = cask transfer trolley; ESD = event sequence diagram; HFE = human failure event; HLW = high-level radioactive waste; IHF = Initial Handling Facility; MAP = mobile access platform; N/A = not applicable; OA = over arching (applies to multiple HFE groups, Section E6.0.2); RC = railcar; SSC = structure, system, or component; SSCs = structures, systems, and components; ST = site transporter; TEV = transport and emplacement vehicle; TT = truck trailer; WP = waste package; WPTT = waste package transfer trolley.

Source: Original

E8 REFERENCES

E8.1 DESIGN INPUTS

The PCSA is based on a snapshot of the design. The reference design documents are appropriately documented as design inputs 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, Sections 3.2.1 and 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 (†).

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.

E8.1.1* AIChE (American Institute of Chemical Engineers) 1992. Guidelines for Hazard Evaluation Procedures. 2nd Edition with Worked Examples. New York, New York: American Institute of Chemical Engineers. TIC: 239050. ISBN: 0-8169-0491-X. (DIRS 103763)

E8.1.2* ASME NOG-1-2004. 2005. *Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder).* New York, New York: American Society of Mechanical Engineers. TIC: 257672. ISBN: 0-7918-2939-1.

E8.1.3* ASME NUM-1-2004. 2005. *Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type).* New York, New York: American Society of Mechanical Engineers. TIC: 259317. ISBN: 0-7918-2938-3.

E8.1.4* ASME RA-S-2002. *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications.* New York, New York: American Society of Mechanical Engineers. TIC: 255508. ISBN: 0-7918-2745-3.

E8.1.5* 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.

E8.1.6† BSC (Bechtel SAIC Company) 2006. *CRCF, IHF, RF, and WHF Canister Transfer Machine Mechanical Equipment Envelope.* 000-MJ0-HTC0-00201-000 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20061120.0011.

E8.1.7* BSC 2006. *Engineering Standard for Repository Component Function Identifiers.* 000-30X-MGR0-00900-000 REV 000. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20060816.0001.

E8.1.8* BSC 2007. *Engineering Standard for Repository Area Codes.* 000-3DS-MGR0-00400-000 REV 004. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070911.0015.

E8.1.9*† BSC 2007. *Repository System Codes.* 000-30X-MGR0-01200-000 REV 00E. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071101.0022.

E8.1.10 BSC 2008. *Initial Handling Facility Event Sequence Development Analysis.* 51A-PSA-IH00-00100-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080207.0005.

E8.1.11* CRA (Corporate Risk Associates) 2006. *A User Manual for the Nuclear Action Reliability Assessment (NARA) Human Error Quantification Technique.* CRA-BEGL-POW-J032, Report No. 2, Issue 5. Leatherhead, England: Corporate Risk Associates. TIC: 259873.

E8.1.12* DOE-STD-1090-2004. *Hoisting and Rigging (Formerly Hoisting and Rigging Manual).* Washington, D.C.: U.S. Department of Energy. ACC: ENG.20060407.0002.

E8.1.13* Dougherty, E.M., Jr. and Fragola, J.R. 1988. Human Reliability Analysis: A Systems Engineering Approach with Nuclear Power Plant Applications. New York, New York: John Wiley & Sons. TIC: 3986. ISBN: 0-471-60614-6. (DIRS 183995)

E8.1.14* Gertman, D.; Blackman, H.; Marble, J.; Byers, J.; and Smith, C. 2005. The SPAR-H Human Reliability Analysis Method. NUREG/CR-6883. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20061103.0009. (DIRS 177326 V)

E8.1.15* Hall, R.E.; Fragola, J.R.; and Wreathall, J. 1982. *Post Event Human Decision Errors: Operator Action Tree/Time Reliability Correlations.* NUREG/CR-3010. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20071220.0211.

E8.1.16* Hamlin, T.L. 2005. *Human Reliability Analysis (HRA) Data Report.* Volume III, Book 2 of *Space Shuttle Probabilistic Risk Assessment*, Rev. 2. Washington, D.C.: National Aeronautics and Space Administration, JSC Safety and Mission Assurance. ACC: MOL.20080311.0023.

E8.1.17* Hannaman, G.W. and Spurgin, A.J. 1984. Systematic Human Action Reliability Procedure (SHARP). EPRI-NP-3583. Palo Alto, California: Electric Power Research Institute. TIC: 252015. (DIRS 157649)

E8.1.18* Hollnagel, E. 1998. *Cognitive Reliability and Error Analysis Method, CREAM.* 1st Edition. New York, New York: Elsevier. TIC: 258889. ISBN: 0-08-0428487

E8.1.19* Lloyd, R.L. 2003. A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002. NUREG-1774. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20050802.0185. (DIRS 174757)

E8.1.20 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. (DIRS 104939)

E8.1.21 NRC 1983. PRA Procedures Guide, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants. NUREG/CR-2300. Two volumes. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 205084. (DIRS 106591)

E8.1.22 NRC 2000. Technical Basis and Implementation Guidelines for a Technique for Human Event Analysis (ATHEANA). NUREG-1624, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 252116. (DIRS 157661)

E8.1.23 NRC 2007. *Preclosure Safety Analysis - Human Reliability Analysis*. HLWRS-ISG-04. Washington, D.C.: Nuclear Regulatory Commission. ACC: MOL.20071211.0230.

E8.1.24* Rasmussen, J. 1983. "Skills, Rules, and Knowledge; Signals, Signs, and Symbols, and Other Distinctions in Human Performance Models." *IEEE Transactions on Systems, Man, and Cybernetics, SMC-13*, (3), 257–266. New York, New York: Institute of Electrical and Electronics Engineers. TIC: 259863.

E8.1.25* Swain, A.D. 1987. *Accident Sequence Evaluation Program Human Reliability Analysis Procedure*. NUREG/CR-4772. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20061103.0026.

E8.1.26* 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.)

E8.1.27* Vesely, W. 2008. "Re: CREAM Errata." E-mail from W.E. Vesely to M. Presley, February, 20, 2008. ACC: MOL.20080220.0081. (DIRS 185075NV)

E8.1.28* Williams, J.C. 1986. "HEART - A Proposed Method for Assessing and Reducing Human Error." *9th Advances in Reliability* Technology *Symposium - 1986.* Bradford, England: University of Bradford. TIC: 259862.

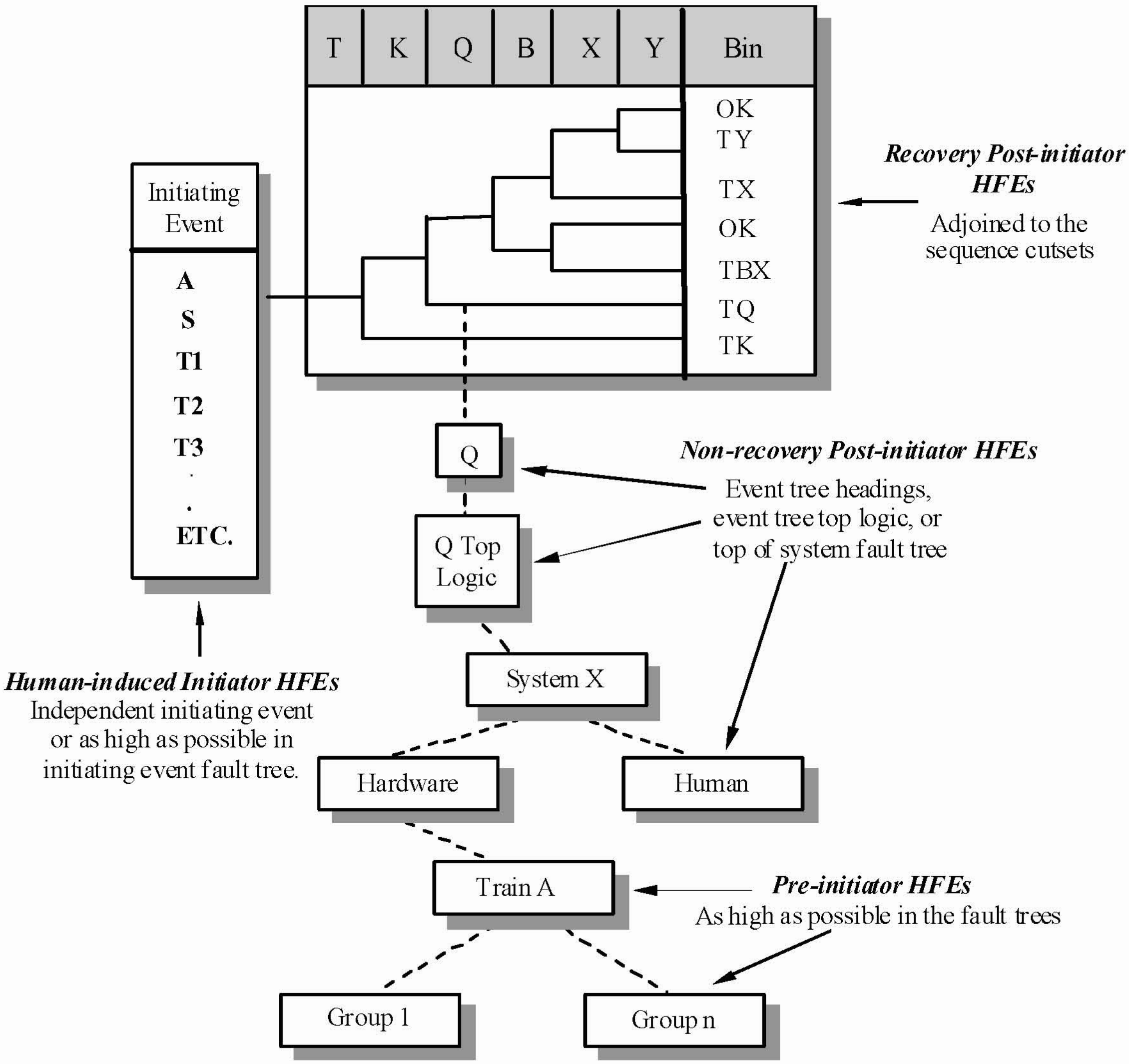
E8.1.29* Williams, J.C. 1988. "A Data-Based Method for Assessing and Reducing Human Error to Improve Operational Performance." *Conference Record for 1988 IEEE Fourth Conference on Human Factors and Power Plants.* Pages 436–450. New York, New York: Institute of Electrical and Electronics Engineers. TIC: 259864.

E8.2 DESIGN CONSTRAINTS

E8.2.1 10 CFR Part 63. 2007. Energy: Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada. U.S. Nuclear Regulatory Commission.

APPENDIX E.I
RECOMMENDED INCORPORATION OF HUMAN FAILURE EVENTS IN THE YMP PCSA

Figure E.I-1 provides a graphical illustration of how HFEs are incorporated into the PCSA.

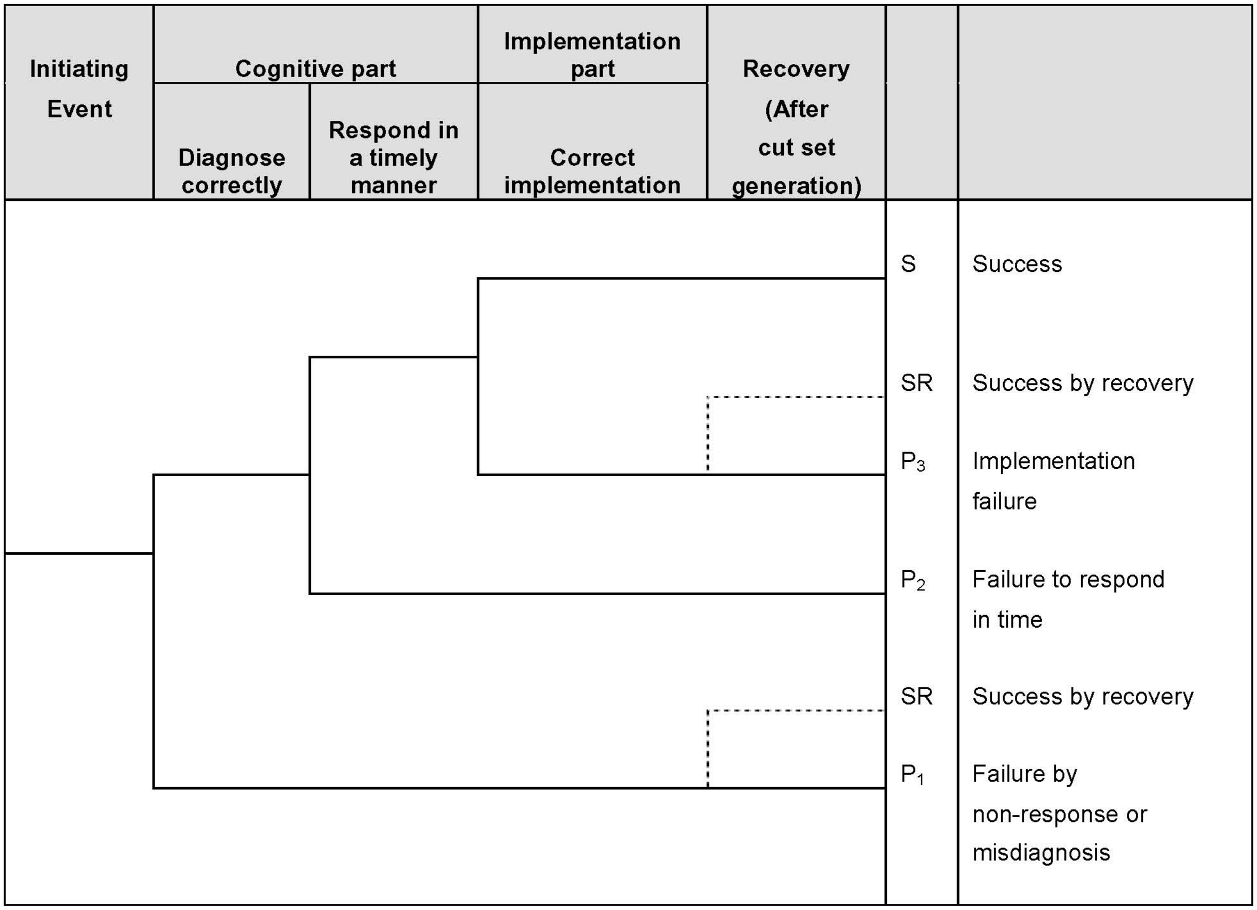


NOTE: HFE = human failure event.

Source: Original

Figure E.I-1. Modeling Strategy for HFE Types

APPENDIX E.II
GENERAL STRUCTURE OF POST-INITIATOR HUMAN ACTIONS



Source: Original

Figure E.II-1 Post Initiator Operator Action Event Tree

The representation in Figure E.II-1 consists of two elements, corresponding to a cognitive part (detection, diagnosis, and decision making) and an implementation (i.e., action) part.

P_1 represents the probability that operators make an incorrect diagnosis and decision and do not realize that they have done so. Some of the reasons for such mistakes are: incorrect interpretation of the procedures, incorrect knowledge of the plant state owing to communication difficulties, and instrumentation problems.

Given that the crew decides what to do correctly, there is still a possibility of failure to respond in time (represented by P_2) or making an error in implementation (represented by P_3).

However, it may be probable in certain scenarios that a recovery action can be taken. This consideration is taken into account after the initial quantification is completed and is applied as appropriate to the dominant cut sets.

APPENDIX E.III
PRELIMINARY (SCREENING) QUANTIFICATION PROCESS FOR HUMAN FAILURE EVENTS

The preliminary quantification process consists of the following:

Step 1—Complete the Initial Conditions Required for Quantification

The preliminary quantification process requires the following:

- The baseline scenarios are available.
- The HFEs and their associated context have been defined.

  – Collect any additional information that is not already collected and that is needed to describe and define the HFEs (and associated contexts).

  – Review all information for clarity, completeness, etc.

  – Interpret and prioritize all information with respect to relevance, credibility, and significance.

Table E.III-1 provides examples of information normally identified using the ATHEANA method (*Technical Basis and Implementation Guidelines for a Technique for Human Event Analysis* (Ref. E8.1.22) that serve as inputs to the quantification process. The HFE/context descriptions in Table E.III-1 touch briefly on the information that is relevant to the screening-level quantification of the HFE. Since the baseline scenario generally touches on much of this information, the point of including the HFE/context descriptions is to summarize the information that pertains to the specific HFE to minimize the need for the analysts to refer back to the baseline scenario, except to obtain additional detail.

Table E.III-1. Examples of Information Useful to HFE Quantification

Information Type	Examples
Facility, conditions, and behavior for possible deviations of the scenarios	Reasonably possible unusual plant behavior and failures of systems; equipment, and indications, especially those that may be unexpected or difficult to detect by operators. Includes presence of interlocks that would have to fail to promote the deviation.
Operating crew characteristics (i.e., crew characterization)	Crew structure, communication style, emphasis on crew discussion of the "big picture."
Features of procedures	Structure, how implemented by operating crews, opportunities for "big picture" assessment and monitoring of critical safety functions, emphasis on relevant issue, priorities, any potential mismatches with deviation scenarios.
Relevant informal rules	Experience, training, practice, ways of doing things - especially those that may conflict with informal rules or otherwise lead operators to take inappropriate actions.
Timing	Plant behavior and requirements for operator intervention versus expected timing of operator response in performing procedure steps, etc.

Table E.III-1. Examples of Information Useful to HFE Quantification (Continued)

Information Type	Examples
Relevant vulnerabilities	Any potential mismatches between the scenarios and expected operator performance with respect to timing, formal and informal rules, biases from operator experience, and training, etc.
Error mechanisms	Any that may be particularly relevant by plant context or implied by vulnerabilities; applicable mechanisms depend upon whether HFE is a slip or mistake. Examples include: failures of attention, possible tunnel vision, conflicts in priorities, biases, missing or misleading indications, complex situations, lack of technical knowledge, timing mismatches and delays, workload, and human–machine interface concerns.
Performance-shaping factors	Those deemed associated with, or triggered by, the relevant plant conditions and error mechanisms.

NOTE: HFE = human failure event.

Source: Original

In Step 1, interpreting and prioritizing all information with respect to relevance, credibility, and significance is especially important if:

- Some information is applicable only to certain scenarios, HFEs, or contexts
- There are conflicts among information sources
- Information is ambiguous, confusing, or incomplete
- Information must be extrapolated, interpolated, etc.

Completion of the "lead-in" initial conditions is primarily performed by a single individual, using the results of the YMP HAZOP evaluation process and reviews of other relevant information sources. Discussions are also held with the Operations Department to augment that information and the resulting write-ups are reviewed by the PCSA facility leads and the HRA team. The initial conditions are refined as part of an open discussion among the experts (in this case, the HRA team for the study) involved in the expert opinion elicitation process. The goal of this discussion is not to achieve a consensus but, rather, to advance the understanding of all the experts through the sharing of distributed knowledge and expertise. In each case, the scenario (or group of similar scenarios) and the HFE in question are described and the vulnerabilities and strong points associated with taking the right action are discussed openly among the HRA team.

Step 2—Identify the Key or Driving Factors of the Scenario Context

The purpose of Step 2 is to identify the key or driving factors on operator behavior/performance for each HFE and associated context. Each expert participating in the elicitation process individually identifies these factors based on the expert's own judgment. Usually, these factors are not formally documented until Step 4.

Typically, there are multiple factors deemed most important to assessing the probability for the HFE in question. This is due to the focus of the ATHEANA search process on combinations of factors that are more likely to result in an integrated context (Ref. E8.1.22). When there is only a single driving factor, it is usually one that is so overwhelming that it alone can easily drive the estimated probability. For example, if the time available is shorter than the time required to perform the actions associated with the HFE, quantification becomes much simpler and other factors need not be considered.

Step 3—Generalize the Context by Matching it With Generic, Contextually-Anchored Rankings, or Ratings

In Step 3, each expert participating in the elicitation process must answer the following question for each HFE: based upon the factors identified in Step 2, how difficult or challenging is this context relative to the HFE being analyzed?

Answering this question involves independent assessments by each expert. In order to perform this assessment, the specifics of the context defined for an HFE must be generalized or characterized. These characterizations or generalizations then must be matched to general categories of failures and associated failure probabilities.

To assist the experts in making their judgments regarding the probability of events, some basic guidance is provided. In thinking about what a particular HEP associated with an HFE may be, they are encouraged to think about similar situations or experiences and use that to help estimate how many times out of 10, 100, 1,000, etc., would they expect crews to commit the HFE, given the identified conditions. The following examples of what different probabilities mean are provided to the experts to help them scale their judgments:

"Likely" to fail (extremely difficult/challenging)	~0.5	(5 out of 10 would fail)
"Infrequently" fails (highly difficult/challenging)[14]	~0.1	(1 out of 10 would fail)
"Unlikely" to fail (somewhat difficult/challenging)	~0.01	(1 out of 100 would fail)
"Highly unlikely" to fail (not difficult/challenging)	~0.001	(1 out of 1000 would fail)

The experts are allowed to select any value to represent the probability of the HFE. That is, other values (e.g., 3E−2, 5E−3) can be used. The qualitative descriptions above are provided initially to give analysts a simple notion of what a particular probability means. For exceptional cases, the quantification approach allows an HEP of 1.0 to be used when failure was deemed essentially certain. The following general guidance in Table E.III-2 is also provided to help calibrate the assessment by providing specific examples that fall into each of the above bins, and is based on the elicited judgment and consensus of the HRA team based on their past experience. This guidance applies to contexts where generally optimal conditions exist during performance of the action. Therefore, the experts should modify these values if they believe that the action may be performed under non-optimal conditions or under extremely favorable conditions. Values may also be adjusted to take credit for design features, controls and interlocks, or procedural safety controls[15,16]. Examples of such adjustments are also provided below; however

[14] The default value is 0.1. This value is used if no preliminary assessment is performed.

[15] As an initial preliminary value, unsafe actions that are backed up by interlocks are assigned a human error probability of 1.0 such that no credit for human performance is taken (i.e., only the interlocks are relied upon to demonstrate 10 CFR Part 63 (Ref. E8.2.1) compliance). If this proves insufficient, a more reasonable preliminary value is assigned to the unsafe action in accordance with this Appendix.

[16] Note that if such credit is taken, then it may be necessary (based on the PCSA results) to include these items in the nuclear safety design basis or the procedural safety controls for the YMP facilities.

these values are not taken to be firm in any sense of the word, but rather simply as examples of where in general terms HEPs may fall and how they may relate to each other. Types of HFEs not listed here can be given values based on being "similar to" HFEs that are listed. Whatever value is selected, the basis is briefly documented.

Table E.III-2. Types of HFEs

PRE-INITIATOR HFEs	
Fail to properly restore a standby system to service	0.1
Failure to properly restore an operating system to service when the degraded state is not easily detectable	0.01
Failure to properly restore an operating system to service when the degraded state is easily detectable	0.001
Calibration error	0.01
HUMAN-INDUCED INITIATOR HFEs	
Failure to properly conduct an operation performed on a daily basis	0.001
Failure to properly conduct an operation performed on a very regular basis (on the order of once/week)	0.01
Failure to properly conduct an operation performed only very infrequently (once/month or less)	0.1
Operation is extremely complex OR conducted under environmental or ergonomic stress	×3
Operation is extremely complex AND conducted under environmental or ergonomic stress	×10
NON-RECOVERY POST-INITIATOR HFEs	
Not trained or proceduralized, time pressure	0.5
Not trained or proceduralized, no time pressure	0.1
Trained and/or proceduralized, time pressure	0.1
Trained and/or proceduralized, no time pressure	0.01

Source: Original

Step 4—Discuss and Justify the Judgments Made in Step 3

In Step 3, each expert independently provides an estimate for each HFE. Once all the expert estimates are recorded, each expert describes the reasons why they chose a particular failure probability. In describing their reasons, each expert identifies what factors (positive and negative) are thought to be key to characterizing the context and how this characterization fit the failure category description and the associated HEP estimate.

After the original elicited estimates are provided, a discussion is held that addresses not only the individual expert estimates but also differences and similarities among the context characterizations, key factors, and failure probability assignments made by all of the experts. This discussion allows the identification of any differences in the technical understanding or interpretation of the HFE versus differences in judgment regarding the assignment of failure probabilities. Examples of factors important to HFE quantification that might be revealed in the discussion include:

- Differences in key factors and their significance, relevance, etc., based upon expert-specific expertise and perspective.

- Differences in interpretations of context descriptions.
- Simplifications made in defining the context.
- Ambiguities and uncertainties in context definitions.

A consensus opinion is not required following the discussion.

Step 5—Refinement of HFEs, associated contexts, and assigned HEPs (if needed)

Based upon the discussion in Step 4, the experts form a consensus on whether or not the HFE definition must be refined or modified, based upon its associated context. If the HFE must be refined or re-defined, this is done in Step 5. If such modifications are necessary, the experts "reestimate" based upon the newly defined context for the HFE (or new HFEs, each with an associated context).

The experts participating in the elicitation process are also allowed to change their estimate after the discussion in Step 4 based on the discussions during that step, whether or not the HFE definition and context are changed. Once again, a consensus is not required.

Step 6—Determine final preliminary HEP for HFE and associated context

The final preliminary value to be incorporated into the PCSA for each HFE is determined in Step 6.

The failure probabilities assigned in the preliminary HRA quantification are based on the context outlined in the base case scenarios and deemed to be "realistically conservative." To help ensure this conservatism, if a consensus value could not be reached, the final failure probability that was assigned to each HFE was determined by choosing the highest assigned probability among the final estimates of the experts participating in the expert elicitation process.

APPENDIX E.IV
SELECTION OF METHODS FOR DETAILED QUANTIFICATION

There are a number of methods available for the detailed quantification of HFEs (preliminary quantification is discussed in Appendix E.III of this analysis). Some are more suited for use for the YMP PCSA than others. A number of methods were considered, but many were rejected as inapplicable or insufficient for use in quantification. Several sources were examined as part of the background analysis for selecting a method for detailed quantification (Ref. E8.1.17; Ref. E8.1.13; Ref. E8.1.24; and Ref. E8.1.21). As discussed in Section E3.2 the following four were chosen:

- ATHEANA expert judgment (Ref. E8.1.22)
- CREAM (Ref. E8.1.18)
- HEART (Ref. E8.1.28)/NARA (Ref. E8.1.11)
- THERP (Ref. E8.1.26).

This appendix discusses the selection process.

Basis for Selection—The selection process was conducted with due consideration of the HRA quantification requirements set forth in the ASME Level 1 PRA standard (Ref. E8.1.4) to the extent that those requirements, which were written for application to NPP PRA, apply to the types of operations conducted at the YMP. Certainly, all of the high level HRA quantification requirements were considered to be applicable. Further, all of the supporting requirements to these high level requirements were considered applicable, at least in regards to their intent. In some cases, the specifics of the supporting requirements are only applicable to NPP HRA and some judgment is needed on how to apply them. This was particularly true of those supporting requirements that judged certain specific quantification methods acceptable. This appendix lays out the specific case for the methods selected for use at the YMP (or, more to the point, the exclusion of certain methods that would normally be considered acceptable under the standard, but are deemed inappropriate for use for the YMP PCSA).

Differences between NPP and **the** YMP Relevant to HRA Quantification—There are a number of contrasts between the operations at the YMP and the operations at an NPP that affect the selection of approaches to performing detailed HRA quantification (Table E.IV-1).

Table E.IV-1. Comparison between NPP and YMP Operations

NPP	YMP
Central control of operations maintained in control room.	Decentralized (local), hands on control for most operations.
Most important human actions are in response to accidents.	Most important human actions are initiating events.
Post-accident response is important and occurs in minutes to hours. Short time response important to model in HRA.	Post-accident response evolves more slowly (hours to days). Short time response not important to model.

Table E.IV-1. Comparison between NPP and YMP Operations (Continued)

NPP	YMP
Multiple standby systems are susceptible to pre-initiator failures.	Standby systems do not play major role in the YMP safeguards, therefore few opportunities for pre-initiator failures.
Auxiliary operators sent by central control room operators to where needed in the plant.	Local control reduces time to respond.
Most actions are controlled by automatic systems.	Most actions are controlled by operators.
Reliance on instrumentation /gauges as operators' "eyes".	Most actions are local, either hands on or televised. Less reliance on man–machine interface.
High complexity of systems, interactions, and phenomena. Actions may be skill, rule, or knowledge based.	Relatively simple process with simple actions. Actions are largely skill based.
Many in operation for decades; HRA may include walk-downs and consultation with operators.	First of a kind; HRA performed for construction application, therefore walk-downs and consultation with operators not feasible.

NOTE: HRA = human reliability analysis; NPP = nuclear power plant; YMP = Yucca Mountain Project.

Source: Original

Assessment of Available Methods—There are essentially four general types of quantification approaches available:

1. Procedure focused methods:

   A. Basis: These methods concentrate on failures that occur during step-by-step tasks (i.e., during the use of written procedures). They are generally based on observations of human performance in the completion of manipulations without much consideration of the root causes or motivations for the performance (e.g., how often does an operator turn a switch to the left instead of to the right).

   B. Methods considered: THERP (Ref. E8.1.26).

   C. Applicability: This method is of limited use for the YMP because important actions are not procedure-driven. Many operations are skill-based and/or semi-automated (e.g., crane operation, trolley operation, CTM operation, TEV operation). However, there are some instances where such an approach would be applicable to certain unsafe actions within an HFE. In addition, the THERP dependency model is adopted by NARA as being appropriate to use within a context-based quantification approach.

   D. Assessment: THERP is retained as an option in the detailed quantification for its dependency model and for limited use when simple, procedure-driven unsafe actions are present within an HFE.

2. Time-response focused methods:

   A. Basis: These methods focus on the time available to perform a task, versus the time required, as the most dominant factor in the probability of failure. They are, for the most part, based on NPP control room observations, studies, and simulator exercises. They also tend to be correlated with short duration simulator exercises (i.e., where there is a clear time pressure in the range of a few minutes to an hour to complete a task in response to a given situation).

   B. As discussed in *Human Reliability Analysis: A Systems Engineering Approach with Nuclear Power Plant Applications* (Ref. E8.1.13), examples of time-response methods include: HCR (Ref. E8.1.13) and TRCs (Ref. E8.1.15).

   C. Applicability: These methods are not applicable to the YMP because most actions do not occur in a control room and, in addition, are generally not subject to time pressure. This is particularly true of the most important HFEs, those that are human-induced initiators. Other than a desire to complete an action in a timely fashion to maintain production schedules, time is irrelevant to these actions, especially in the context of the type of time pressure considered by these methods. Even those actions at the YMP that may take place in a control room in response to an event sequence and have time as a factor would only require response in the range of hours or days, which is outside the credible range for these methods.

   D. Assessment: No use can be identified for these methods within the YMP PCSA. None of them are retained.

3. Context and/or cognition driven methods:

   A. Basis: These methods focus on the context and motivations behind human performance rather than the specifics of the actions, and as such are independent of the specific facility and process. To the extent that some of the methods are data-driven (i.e., they collect and use observations of human performance) the data utilized is categorized by GTT rather than by the type of facility or equipment where the human failure occurred. This makes them more broadly applicable to various industries, tasks, and situations, in large part because they allow context-specific PSFs to be considered. This allows for them to support a variety of contexts, individual performance factors (e.g., via PSFs) and human factor approaches.

   B. Methods considered: HEART (Ref. E8.1.28 and Ref. E8.1.29)/NARA (Ref. E8.1.11), CREAM (Ref. E8.1.18), and ATHEANA (Ref. E8.1.22) expert judgment.

   C. Applicability: The broad applicability of these methods and their flexibility of application make them most suited for application at the YMP. The use of information from a broad range of facilities and other performance regimes (e.g., driving, flying) support their use as facility-independent methods. The generic tasks considered can be applied to the types of actions of most concern to the YMP (i.e., human-induced initiators) as opposed to the more narrow definitions used in other approaches that make it difficult to use them for other than post-initiator or pre-initiator actions.

D. Assessment: Optimally it would be convenient to use only one of the three methods of this type for all the detailed quantification. However, HEART (Ref. E8.1.28)/NARA (Ref. E8.1.11) and CREAM (Ref. E8.1.18) approach their GTTs slightly differently and also use different PSFs and adjustment factors. There are unsafe actions within the YMP HFEs that would best fit the HEART (Ref. E8.1.28)/NARA (Ref. E8.1.11) approach and others that would best fit the CREAM (Ref. E8.1.18) approach. In addition, the union of the two approaches still has some gaps that would not cover a small subset of unsafe actions for the YMP (primarily in the area of unusual acts of commission). One gap relates to dependencies between actions, but in this case NARA (Ref. E8.1.11) specifically endorses the THERP (Ref. E8.1.26) approach and so this is used. However, other gaps exist. For these cases, the ATHEANA (Ref. E8.1.22) expert judgment approach provides a viable and structured framework for the use of judgment to establish the appropriate HEP values in a manner that would meet the requirements of the ASME RA-S-2002 (Ref. E8.1.4) standard. Therefore, all three of these methods are retained for use and the selection of one versus the other is made based on the specific unsafe action being quantified. This is documented as appropriate in the actual detailed quantification of each HFE.

4. Simplified methods:

A. Basis: These methods use the results of past PRAs to focus attention on those HFEs that have dominated risk. These are essentially PRA results from NPPs. As such, they pre-suppose NPP situations and actions, and define important PSFs based on these past NPP PRAs. They have very limited (if any) ability to investigate context, individual and human factors that are beyond NPP experience. The HEPs that result from applying these methods are calibrated to other NPP methods.

B. Methods considered: ASEP (Ref. E8.1.25), SPAR-H (Ref. E8.1.14).

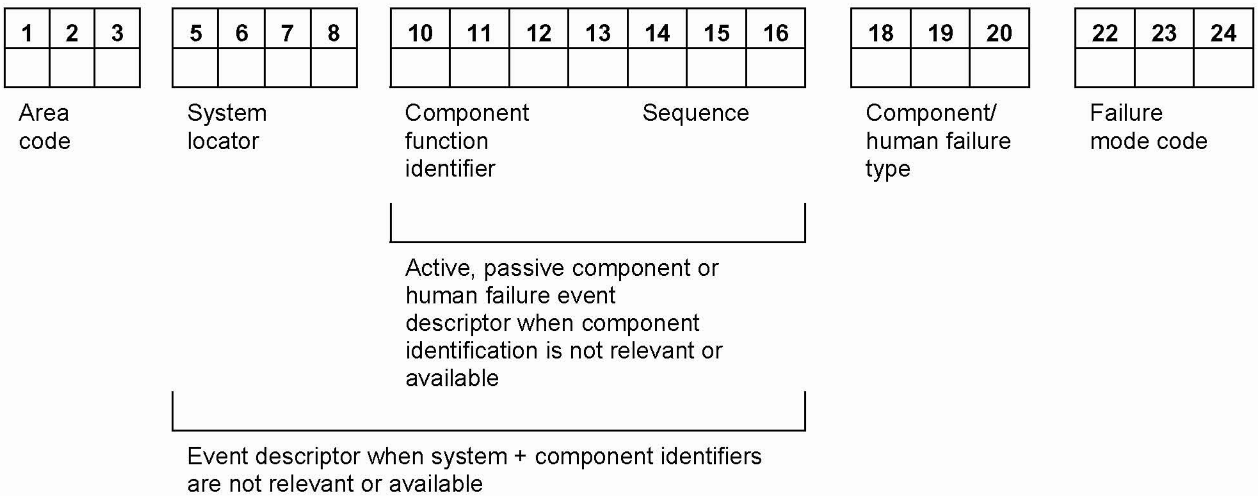
C. Applicability: These methods are clearly biased by their very close dependence on the results of past NPP PRAs. They are too limited for application beyond the NPP environment. They are not simply inappropriate for this application, but it would be extremely difficult to make a sound technical case regarding technical validity.

D. Assessment: No use can be identified for these methods within the YMP PCSA or any technical case made supporting them for a non-NPP application. None of them are retained.

APPENDIX E.V
HUMAN FAILURE EVENTS NAMING CONVENTION

Event names for HFEs in the YMP PCSA model follow the general structure of the naming convention for fault tree basic events. This is true whether the HFE is modeled in a fault tree, directly on an event tree, or as an initiating event. The convention, as adapted for HFEs, is as follows:

This basic event naming convention in Figure E.V-1 below is provided to ensure consistency with project standards and to permit this information to fit into a 24-character SAPHIRE field such that each basic event can be correlated to a unique component or human failure.



Source: Original

Figure E.V-1. Basic Event Naming Convention

The area code defines the physical design or construction areas where a component would be installed. Area codes are listed in *Engineering Standard for Repository Area Codes,* (Ref. E8.1.8). These codes are used rather than the facility acronyms to maintain consistency with Engineering. In this system, the Canister Receipt and Closure Facility is designated by area code 060, the Wet Handling Facility is 050, the Receipt Facility is 200, the IHF is 51A, and Subsurface is 800. Intra-Site Operations could fall under one of several repository area codes and therefore the most appropriate code to use was the repository general area code. However, this code was insufficient for the purposes of this analysis, and a designator of ISO was substituted instead. For the majority of cases, the area coding of HFEs in Attachment E reflects the location of the operations being evaluated, such as ISO for Intra-Site Operations. However, for certain HFEs, the coding corresponds to the location of the systems impacted by the human failure, such as HVAC, which is specific to the CRCF and therefore retains the 060 coding, and AC power, which retains the 26x and 27x coding. For these specific instances, such coding provides better traceability of the HFE back to the affected equipment.

The system locator code identifies operational systems and processes. System locator codes (four characters) are listed in Table 1 of *Repository System Codes* (Ref. E8.1.9). These are generally three or four characters long, such as VCT for tertiary confinement HVAC.

The component function identifiers identify the component function and are listed in the *Engineering Standard for Repository Component Function Identifiers* (Ref. E8.1.7). These are generally three or four characters long. Some BSC component function identifiers for typical components are shown in Table E.V-1, but in cases where there is not an equivalent match, the most appropriate PCSA type code should be used (also given in Table E.V-1).

The sequence code is a numeric sequence and train assignment (suffix), if appropriate, that uniquely identifies components within the same area, system, and component function.

If an HFE is related to the failure of an individual component with an existing component function identifier and sequence code, the naming scheme should utilize these codes in the event name. If an HFE is such that these codes do not apply, the basic event name can be a free form field for describing the nature of the event, such as HCSKSCF for operator topples cask during scaffold movement or HFCANLIDAJAR for operator leaves canister lid ajar, utilizing either seven characters when there is a relevant system locator code, or 12 characters when no system codes are applicable.

The human failure type and failure mode codes are three characters each, consistent with the coding provided in Table E.V-1 below.

For HFEs, the type code always begins with HF and continues with a one letter designator for the HFE temporal phase: P for pre-initiator, I for human-induced initiator, N for non-recovery post-initiator, R for recovery post-initiator (this latter code is not used during preliminary analysis).

Table E.V-1. Human Failure Event Type Codes and Failure Mode Codes

PRE-INITIATOR HFEs; TYP=HFP		FMC=
Fail to properly restore a standby system to service		RSS
Failure to properly restore an operating system to service when the degraded state is not easily detectable		ROH
Failure to properly restore an operating system to service when the degraded state is easily detectable		ROE
Calibration error		CAL
HUMAN-INDUCED INITIATOR HFEs; TYP=HFI		
Failure to properly conduct an operation	Operation is performed on a daily basis.	NOD
	Operation is performed on a very regular basis (on the order of once per week)	NOW
	Operation is performed only very infrequently (once per month or less)	NOM
Operation is extremely complex OR conducted under environmental or ergonomic stress	Operation is performed on a daily basis.	COD
	Operation is performed on a very regular basis (on the order of once per week)	COW
	Operation is performed only very infrequently (once per month or less)	COM

Table E.V-1. Human Failure Event Type Codes and Failure Mode Codes (Continued)

PRE-INITIATOR HFEs; TYP=HFP		**FMC=**
Operation is extremely complex AND conducted under environmental or ergonomic stress	Operation is performed on a daily basis.	CSD
	Operation is performed on a very regular basis (on the order of once per week)	CSW
	Operation is performed only very infrequently (once per month or less)	CSM
NON-RECOVERY POST-INITIATOR HFEs; TYP=HFN		
Not trained or proceduralized, time pressure		NPT
Not trained or proceduralized, no time pressure		NPN
Trained and/or proceduralized, time pressure		TPT
Trained and/or proceduralized, no time pressure		TPN
RECOVERY POST-INITIATOR HFEs; TYP=HFR		
Not trained or proceduralized, time pressure		NPT
Not trained or proceduralized, no time pressure		NPN
Trained and/or proceduralized, time pressure		TPT
Trained and/or proceduralized, no time pressure		TPN

NOTE: FMC = failure mode code; HFE = human failure event; HFI = human-induced initiator HFE; HFN = human failure non-recovery post-initiator HFE; HFP = pre-initiator HFE; HFR = human failure recovery post-initiator HFE; TYP = type.

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ATTACHMENT F
FIRE ANALYSIS

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ACRONYMS AND ABBREVIATIONS

Acronyms

CTT	cask transfer trolley
HEPA	high-efficiency particulate air
HLW	high-level radioactive waste
HVAC	heating, ventilation, and air conditioning
IHF	Initial Handling Facility
MCC	motor control center
NFPA	National Fire Protection Association
NSNF	naval spent nuclear fuel
PCSA	preclosure safety analysis
WPTT	waste package transfer trolley
YMP	Yucca Mountain Project

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F1 INTRODUCTION

This document describes the work scope, definitions and terms, method, and results for the fire analysis performed as part of the Yucca Mountain Project (YMP) preclosure safety analysis (PCSA). Fire analysis is divided into four major areas:

1. Initiating event identification
2. Initiating event quantification (including both ignition frequency and propagation probability)
3. Fragility analysis (including convolution of fragility and hazard curves)
4. Fire analysis model development and quantification.

Within the task, the internal events PCSA model is evaluated, with respect to fire initiating events, and modified as necessary to address fire-induced failures that lead to exposures. The lists of fire-induced failures that are included in the model are evaluated as to fire vulnerability, and fragility analyses are conducted as needed. All calculations are performed in Excel and included in Attachment H in *IHF Fire Frequency - no suppression.xls* and *IHF CB Report.xls.*

F2 REFERENCES: DESIGN INPUTS

The PCSA is based on a snapshot of the design. The reference design documents are appropriately documented as design inputs 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, Sections 3.2.1 and 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 (†)..

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.

F2.1 ANSI/ANS-58.23-2007. *Fire PRA Methodology.* La Grange Park, Illinois: American Nuclear Society. TIC: 259894.

F2.2 ASME RA-S-2002. *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*. New York, New York: American Society of Mechanical Engineers. TIC: 255508. ISBN: 0-7918-2745-3.

F2.3† BSC (Bechtel SAIC Company) 2007. *Equipment Motor Horsepower and Electrical Requirements Analysis*. 000-M0A-H000-00100-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070816.0001.

F2.4*† BSC 2007. *Initial Handling Facility 480 V Load Center 51A-EEN0-LC-00001 Single Line Diagram*. 51A-E10-EEN0-00101-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0011.

F2.5*† BSC 2007. *Initial Handling Facility 480 V Load Center 51A-EEN0-LC-00002 Single Line Diagram*. 51A-E10-EEN0-00501-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0015.

F2.6*† BSC 2007. *Initial Handling Facility 480 V Load Center 51A-EEN0-LC-00003 Single Line Diagram*. 51A-E10-EEN0-00901-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0019.

F2.7* BSC 2007. *Initial Handling Facility 480V MCC 51A-EEN0-MCC-00001 Single Line Diagram*. 51A-E10-EEN0-00201-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0012.

F2.8*† BSC 2007. *Initial Handling Facility 480 V MCC 51A-EEN0-MCC-00002 Single Line Diagram*. 51A-E10-EEN0-00301-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0013.

F2.9* BSC 2007. *Initial Handling Facility 480 V MCC 51A-EEN0-MCC-00003 Single Line Diagram*. 51A-E10-EEN0-00401-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0014.

F2.10*† BSC 2007. *Initial Handling Facility 480 V MCC 51A-EEN0-MCC-00004 Single Line Diagram*. 51A-E10-EEN0-00601-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0016.

F2.11*† BSC 2007. *Initial Handling Facility 480 V MCC 51A-EEN0-MCC-00005 Single Line Diagram*. 51A-E10-EEN0-00701-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0017.

F2.12* BSC 2007. *Initial Handling Facility 480 V MCC 51A-EEN0-MCC-00006 Single Line Diagram*. 51A-E10-EEN0-00801-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0018.

F2.13*† BSC 2007. *Initial Handling Facility 480 V MCC 51A-EEN0-MCC-00007 Single Line Diagram*. 51A-E10-EEN0-01001-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0020.

F2.14* BSC 2007. *Initial Handling Facility Cask Cavity Gas Sampling System Piping & Instrument Diagram*. 51A-M60-MRE0-00101-000 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070328.0010.

F2.15*† BSC 2007. *Initial Handling Facility Chilled Water System P&ID*. 51A-M60-PSC0-00101-000 REV A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071119.0017.

F2.16* BSC 2007. *Initial Handling Facility Composite Vent Flow Diagram Non-Confinement HVAC Systems*. 51A-M50-VNI0-00101-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0003.

F2.17* BSC 2007. *Initial Handling Facility Composite Vent Flow Diagram Tertiary Confinement HVAC Miscellaneous Areas*. 51A-M50-VCT0-00201-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0002.

F2.18* BSC 2007. *Initial Handling Facility Composite Vent Flow Diagram Tertiary Confinement HVAC Supply & Exhaust Systems*. 51A-M50-VCT0-00101-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0001.

F2.19* BSC 2007. *Initial Handling Facility Confinement Areas HEPA Exhaust System Ventilation & Instrumentation Diagram*. 51A-M80-VCT0-00102-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0005.

F2.20* BSC 2007. *Initial Handling Facility Confinement Areas HVAC Supply System Ventilation & Instrumentation Diagram*. 51A-M80-VCT0-00101-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0004.

F2.21* BSC 2007. *Initial Handling Facility Confinement Battery Room HEPA Exhaust System Ventilation & Instrumentation Diagram*. 51A-M80-VCT0-00202-000 REV 00B. Las Vegas, Nevada, Bechtel SAIC Company. ACC: ENG.20080102.0009.

F2.22* BSC 2007. *Initial Handling Facility Confinement Cask Prep Area and WP Loadout Rm HVAC System Ventilation & Instrumentation Diagram*. 51A-M80-VCT0-00203-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0010.

F2.23* BSC 2007. *Initial Handling Facility Confinement Electrical and Battery Room HVAC System Ventilation & Instrumentation Diagram*. 51A-M80-VCT0-00201-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0008.

F2.24* BSC 2007. *Initial Handling Facility Electrical Room Equipment Layout*. 51A-E40-EEN0-00101-000 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20070521.0003.

F2.25 BSC 2007. *Initial Handling Facility General Arrangement Ground Floor Plan*. 51A-P10-IH00-00102-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071226.0017.

F2.26 BSC 2007. *Initial Handling Facility General Arrangement Plan at Elevation +73'-0"*. 51A-P10-IH00-00104-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071226.0019.

F2.27 BSC 2007. *Initial Handling Facility General Arrangement Second Floor Plan*. 51A-P10-IH00-00103-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071226.0018.

F2.28*† BSC 2007. *Initial Handling Facility Hot Water System P&ID*. 51A-M60-PSH0-00101-000 REV 00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071119.0019.

F2.29* BSC 2007. *Initial Handling Facility Non-Confinement Areas Air Distribution System (North) Ventilation & Instrumentation Diagram*. 51A-M80-VNI0-00102-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0012.

F2.30* BSC 2007. *Initial Handling Facility Non-Confinement Areas HVAC Supply System Ventilation & Instrumentation Diagram*. 51A-M80-VNI0-00101-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0011.

F2.31* BSC 2007. *Initial Handling Facility Non-Confinement Operations Area HVAC System Ventilation & Instrumentation Diagram*. 51A-M80-VNI0-00104-000 REV 00B. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080102.0014.

F2.32* BSC 2007. *Initial Handling Facility UPS 51A-EEP0-UJX-00002 Single Line Diagram*. 51A-E10-EEP0-00201-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0022.

F2.33* BSC 2007. *Initial Handling Facility UPS 51A-EEP0-UJX-00001 Single Line Diagram*. 51A-E10-EEP0-00101-000 REV 00C. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071201.0021.

F2.34* BSC (Bechtel SAIC Company) 2007. *Preliminary Throughput Study for the Initial Handling Facility*. 51A-30R-IH00-00100-000-001. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20071102.0021.

F2.35* EPRI (Electric Power Research Institute) and NRC (U.S. Nuclear Regulatory Commission) 2005. *Detailed Methodology*. Volume 2 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*. EPRI TR-1011989 and NUREG/CR-6850. Palo Alto, California: Electric Power Research Institute. ACC: MOL.20070323.0062.

F2.36* EPRI and NRC 2005. *Summary & Overview*. Volume 1 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*. EPRI-1011989 and NUREG/CR-6850. Palo Alto, California: Electric Power Research Institute. ACC: MOL.20070323.0061.

F2.37 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.

F2.38 Ahrens, M. 2007. *Structure Fires in Radioactive Material Working Facilities and Nuclear Energy Plants of Non-Combustible Construction, 1980-1998*. Quincy, Massachusetts: National Fire Protection Association. TIC: 259983.

F2.39* SAIC (Science Applications International Corporation) 2002. *Chemical Agent Disposal Facility Fire Hazard Assessment Methodology.* SAIC-01/2650. Abingdon, Maryland: Science Applications International Corporation. ACC: MOL.20080115.0138.

F2.40* Tillander, K. 2004. *Utilisation of Statistics to Assess Fire Risks in Buildings*. PhD dissertation. Espoo, Finland: VTT Technical Research Centre of Finland. TIC: 259928. ISBN: 951-38-6392-1.

F2.41* Winkler, R. L., and Hays, W. L. 1975. *Statistics: Probability, Inference, and Decision*. Series in Quantitative Methods for Decision Making. 2nd Edition. Winkler, R.L., ed., New York, New York: Holt, Rinehart, and Winston. TIC: 259976. ISBN: 0-03-014011-0.

F3 BOUNDARY CONDITIONS

F3.1 INTRODUCTION

The general boundary conditions used during the analysis of fire vulnerabilities and fire model development are clearly stated and documented. In general, the boundary conditions are compatible with those usually applied to fire events. The principal boundary conditions for the fire analysis are listed in the following sections.

F3.2 PLANT OPERATIONAL STATE

The initial state of the facility is normal, with each system operating within its limiting condition of operation.

F3.3 CREDIT FOR AUTOMATIC FIRE SUPPRESSION SYSTEMS

The automatic fire suppression systems, although designed to meet all requirements and standards for fire suppression systems in nuclear facilities, are considered not important to safety, and therefore no credit is taken for their operation.

F3.4 NUMBER OF FIRE EVENTS TO OCCUR

The facility is analyzed to respond to one fire event at a given time. Additional fire events as a result of independent causes or of re-ignition once a fire is extinguished are not considered.

F3.5 IGNITION SOURCE COUNTING

Ignition sources are counted in accordance with applicable counting guidance contained in NUREG/CR-6850 (*Detailed Methodology*, Volume 2 of *EPRI/NRC-RES Fire PRA Methodology* *for Nuclear Power Facilities* (Ref. F2.35)) and *Summary & Overview*, Volume 1 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.36).

F3.6 FIRE CABLE AND CIRCUIT FAILURE ANALYSIS

Unlike nuclear power plants, which depend on the continued operation of equipment to prevent fuel damage, the YMP facilities cease operating upon a loss of power or control. Therefore, fire damage in rooms that do not contain waste cannot result in an increased level of radiological exposure. Cable and circuit analysis in these rooms is not required.

F3.7 HEATING, VENTILATION, AND AIR CONDITIONING FIRE ANALYSIS

Heating, ventilation, and air conditioning (HVAC) is not relied upon to mitigate potential releases associated with large fire event sequences. In recognition of a large amount of fire generated, non-radiological particulates could render the HVAC filters ineffective. HVAC can be credited for localized fires unless HVAC control or power circuits are present in the area of the fire.

F3.8 NO OTHER SIMULTANEOUS INITIATING EVENTS

It is standard practice to not consider the occurrence of other initiating events (human-induced and naturally occurring) during the time span of an event sequence because (1) the probability of two simultaneous initiating events within the time span is small, and (2) each initiating event would cease operations of the Initial Handling Facility (IHF), which further reduces the conditional probability of the occurrence of a second initiating event, given that the first has occurred.

F3.9 DATA COLLECTION SCOPE

The fire ignition data collection and analysis are performed for locations relevant to waste handling in the facilities.

F3.10 COMPONENT FAILURE MODES

The failure mode of a structure, system, or component affected by a fire is the most severe with respect to consequences. For example, the failure mode for a canister could be the overpressurization of a reduced-strength canister.

F3.11 COMPONENT FAILURE PROBABILITY

Fires large enough to fail waste containment components would be large enough to fail all active components in the same room. Active components fail in a de-energized state for such fires.

F3.12 INTERNAL EVENTS PRECLOSURE SAFETY ANALYSIS MODEL

To implement the systems analysis guidance contained herein, the fire PCSA team uses the internal events PCSA model, which is developed concurrently with the fire PCSA. This internal events PCSA is used as the basis for the fire PCSA. The internal events PCSA is in general conformance with the ASME PRA *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications* (Ref. F2.2).

F4 ANALYSIS METHOD

F4.1 INTRODUCTION

Nuclear power plant fire risk assessment techniques, as discussed in the following sections, have limited applicability to facilities such as the IHF or other facilities in the geologic repository operations area. The general methodological basis of this analysis is the *Chemical Agent Disposal Facility Fire Hazard Assessment Methodology* (Ref. F2.39), which applies to facilities that are similar to the geologic repository operations area in that they are handling and disposal facilities for highly hazardous materials. This approach is data based, in that it uses actual fire ignition and fire propagation experience to determine fire initiating event frequencies. That approach has been adapted to use data applicable to the YMP waste handling facilities. To the extent applicable to a nonreactor facility, NUREG/CR-6850, *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volumes 1 and 2* (Ref. F2.36 and Ref. F2.35) are also considered in the development of this analysis method. The method complies with the applicable requirements of the ANS *Fire PRA Methodology* (Ref. F2.1) that are relevant to a nonreactor facility. Many of the definitions, modeling approximations, and requirements of these documents were used to develop this document.

F4.2 IDENTIFICATION OF INITIATING EVENTS

Current techniques in fire risk assessment for nuclear power plants focus on fire that can damage electrical and control circuits or impact other equipment that can compromise process and safety systems. This type of approach is not generally applicable to the YMP because loss of electric power is a safe state, except for the need for HVAC after a release of radionuclides. In general, when systems are affected by fire, they cease to function. While at a nuclear power plant fire is of concern, at the YMP fire means that fuel handling stops, and initiating events capable of producing elevated levels of radioactivity are essentially unrealizable. While it is theoretically possible that a fire could inadvertently result in a drop of a cask or canister, it is difficult (if not impossible) to identify any mechanisms by which this action would occur due to fire that would not be much more likely to occur by other means. Of much greater concern at the YMP is the potential for a fire to directly affect the waste containers and cause a breach that would result in a release. The fire analysis, therefore, focuses on the potential for a fire to directly affect the waste containers and cause a breach that would result in a release, rather than analyzing fires that would remove power from fuel handling systems. After a release of radionuclides, the HVAC system, with its high-efficiency particulate air (HEPA) filtration, aids in the abatement of radioactivity that is released from buildings. However, the occurrence of fires tends to significantly reduce the effectiveness of HEPA filtration; the fire event sequence analysis, therefore, does not rely on this system. Consideration is given both to fires that start in rooms containing waste and fires that start in other rooms and propagate to where waste is located. The steps of this process are provided in the following sections.

F4.2.1 Identify Fire-Rated Barriers and Designate Fire Zones

The facility is broken into fire zones based on the location of fire-rated barriers. The rating of the barriers is not significant to the methodology, so all rated barriers are considered. In order for a fire zone to exist, the penetrations, doorways, and ducts must also be limited to the perimeter of the zone. It should be noted that a floor is always considered to be a fire barrier as long as it is solid. Zones are identified by a number, determined by the analyst, and consist of one or more rooms.

F4.2.2 Identify the Rooms Where Waste Can Be Present

Each room where waste can be present, even if only for a brief time, is listed. The first set of fire initiating events to be considered in the PCSA is fires that affect each of these rooms but do not affect other rooms that could contain waste.

F4.2.3 Define Local Initiating Events

Fire ignition occurrences are identified for each room within a fire zone. The total occurrences of a fire within a room containing a waste form are composed of the occurrences of ignitions in that room plus the occurrences of ignitions in surrounding rooms within the fire zone, which propagate across room boundaries to the room containing the waste form. The locations of fire initiating events were identified in the master logic diagram.

F4.2.4 Define Large Fire Initiating Events

Traditional fire risk studies for nuclear power plants have tended to ignore large fires, arguing that the fire barriers in place would prevent such occurrences. However, actual observed historical data shows that large fires in buildings occur. Large fires are defined for this study as those that spread to encompass the entire building, as recognized in the latest fire risk guidance from *Detailed Methodology* Volume 2 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* ((Refs. F2.35) and *Summary & Overview* (Ref. F2.36) Volume 1, Section 11.5.4, for example). There, potential large fire initiating events are identified. The general approach follows.

Except during the short time that waste forms are being lifted by a canister transfer machine, in the YMP facilities, waste forms are located on the ground floor. Continuing with the focus on rooms that contain waste forms, large fires may be divided two ways. One is associated with fires that start on the ground floor and spread to the entire building. The other is a fire that starts anywhere else in the building and spreads to the entire building.

As a practical analysis technique, any fire that spreads out of a fire area is considered a large fire.

F4.3 QUANTIFICATION OF FIRE IGNITION FREQUENCY

The quantification of initiating event frequency involves three steps. First, the overall frequency of fire ignition for the facility is determined, and then that frequency is allocated to the individual room in the facility, based on the number and types of ignition sources in the rooms. Types of ignition sources are characterized in general terms (e.g., mechanical, electrical, combustible liquid). Finally, propagation probabilities are applied to determine the overall frequency that a fire reaches the area of the waste. Quantification uses data from the following sources for equipment ignition frequencies and conditional probabilities of propagation:

- *Detailed Methodology*, Volume 2 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.35)
- *Summary & Overview*, Volume 1 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.36)
- *Fires in or at Industrial Chemical, Hazardous Chemical, and Plastic Manufacturing Facilities: 1988 - 1997 Unallocated Annual Averages and Narratives* (Ref. F2.37)
- *Structure Fires in Radioactive Material Working Facilities and Nuclear Energy Plants of Non-Combustible Construction, 1980-1998* (Ref. F2.38)
- *Chemical Agent Disposal Facility Fire Hazard Assessment Methodology* (Ref. F2.39)
- *Utilisation of Statistics to Assess Fire Risks in Buildings* (Ref. F2.40).

F4.3.1 Determine the Overall Facility Fire Frequency

There is insufficient data available regarding the total frequency of fires in facilities comparable to the YMP. NUREG/CR-6850 (Volume 2 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.35) and Volume 1 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.36)) provides an overall frequency for a typical nuclear power plant, but these plants are much larger and more complex than the YMP facilities. Therefore, it has been decided that a more generic fire ignition frequency approach be used that relates building size to total fire frequency for various broad categories of facilities (*Utilisation of Statistics to Assess Fire Risks in Buildings* (Ref. F2.40)). This approach applies the following equation to overall fire ignition frequency.

Determine the Fire Frequency per Unit Area—The frequency per unit area is expressed by the following equation:

$$f_m(A) = c_1A^r + c_2A^s \quad \text{(Eq. F-1)}$$

where f_m is the fire ignition frequency per square meter per year, A is the floor area (in square meters) and c_1, c_2, r, and s are coefficients that were determined from historical data observations for different types of facilities.

For industrial buildings, the parameter values are as follows:

$$c_1 = 3 \times 10^{-4}; c_2 = 5 \times 10^{-6}; r = \text{-}0.61; \text{ and } s = \text{-}0.05 \quad \text{(Eq. F-2)}$$

Equation F-1 relates the frequency per unit area to the total area of the facility. This correlation was determined from the historical data, which showed that total fire frequency was not linearly related to the size of the facility. Rather, the frequency per unit area was affected by the size of the facility, and the larger the facility, the lower the frequency per unit area.

Determine the Total Fire Frequency for the Facility—The total frequency of fire ignition for the building is thus represented by the following equation:

$$f_{fire} = f_m(A) * A \quad \text{(Eq. F-3)}$$

F4.3.2 Determine the Fire Ignition Frequency in Each Room

The approach to allocating the fire ignition frequency is based on the approach used in *Detailed Methodology,* Volume 2 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.35), and *Summary & Overview,* Volume 1 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.36), and *Chemical Agent Disposal Facility Fire Hazard Assessment Methodology* (Ref. F2.39). Both of these approaches determine the fraction of the total facility ignition frequency associated with various categories of equipment (i.e., ignition source category), then determine a facility-specific ignition frequency for each piece of equipment in each category, and then determine the total ignition frequency in the room based on the ignition source population in the room.

F4.3.2.1 Fraction of Fire Ignition Frequency Associated with Each Ignition Source Category

Detailed Methodology, Volume 2 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Refs.F2.35) and *Summary & Overview,* Volume 1 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.36) have data for these fractions for nuclear power plants, and *Chemical Agent Disposal Facility Fire Hazard Assessment Methodology* (Ref. F2.39) has data for these frequencies for chemical process plants. Neither of these data sets is the best for the facilities at the YMP. Therefore, the National Fire Protection Association (NFPA) was requested to provide an analysis, *Structure Fires in Radioactive Material Working Facilities and Nuclear Energy Plants of Non-Combustible Construction, 1980-1998* (Ref. F2.38) of the data in their proprietary database on the distribution of fires by equipment type in all nuclear facilities of noncombustible construction. NFPA distinguishes between a large number of equipment types that can cause ignition of a fire. There is an insufficient amount of data to justify retaining this number of equipment types, so the equipment types were consolidated into a set of ignition source categories. These categories are defined in Appendix F.I.

Using the data by category, an analysis is performed to determine the fraction of fires that are caused by each category. That analysis is documented in Appendix F.II.

The total fire ignition frequency from Section F4.3.1 is multiplied by each of these factors to determine the total fire ignition frequency due to each equipment type. For example, the total ignition frequency due to electrical equipment for a given facility follows:

$$f_{elec\text{-}all} = f_{fire} * 0.086 \quad \text{(Eq. F-4)}$$

F4.3.2.2 Individual Ignition Source Fire Ignition Frequency

The next step is to determine the fire ignition frequency from each piece of equipment in each category. As is done in *Detailed Methodology,* Volume 2 of *EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities* (Ref. F2.35) and *Summary & Overview,* Volume 1 of