RAI Volume 2, Chapter 2.1.1.5, First Set, Number 4:

> Provide information on potential worker dose resulting from handling of individual spent nuclear fuel assemblies with damaged cladding in the WHF pool, during normal operations. Alternatively, provide a rationale why this type of exposure (e.g., indoor exposure of workers to airborne radioactive material) does not need to be considered in the DOE consequence analysis (SAR Section 1.8).

1. RESPONSE

Potential worker airborne dose contributions from damaged fuel assemblies are included in the preclosure consequence analysis for normal operations as reported in SAR Table 1.8-32. However, these contributions occur during cutting open dual-purpose canisters (DPCs) and preparing transportation casks for lid removal prior to any spent fuel assembly handling operations within the pool. Based on historical industry experience and the Wet Handling Facility (WHF) design features, incremental contribution of gaseous releases during WHF pool fuel handling operations would be insignificant.

Potential airborne releases from damaged fuel assemblies in the DPCs and casks received at the WHF are confined and eventually vented through the heating, ventilation, and air-conditioning system during preparation operations. The cask and DPC cooling system in the WHF cools the spent nuclear fuel inside the cask or DPC by injecting borated pool water into the cask or DPC prior to transfer into the pool. The steam created by evaporative cooling is confined to the cask or DPC and is directed out through vent piping. This piping discharges into a gas/liquid separator. Any condensate that could also have radioactive particulates is separated from the steam and routed to the pool water treatment suction piping. Gaseous releases are routed directly to the WHF heating, ventilation, and air-conditioning system passing through high-efficiency particulate air (HEPA) filters prior to discharge to the outside environment (SAR Section 1.2.5.2.3.2). There are no unfiltered releases to the environment or releases directly into the interior of the WHF during this operation. As noted in SAR Section 1.8.4.1.1, the WHF releases that pass through the HEPA filters to the outside environment are then modeled conservatively as reentering the WHF ventilation system intake. Potential worker doses within the WHF from airborne releases include this contribution. The highest airborne dose from these releases is 15 mrem/year for a worker in the WHF (SAR Table 1.8-32).

Fuel cladding damage could occur during handling or storage at originator sites, shipping to the repository, or cooling operations. For fuel cladding damage larger than pinhole leaks and hairline cracks including failed fuel in canisters, any remaining fission product gases and volatile species are expected to be released into the cask or DPC and then discharged through vent piping and HEPA filters during the aforementioned cask cooling process. Any fuel particulates that may escape through the fuel cladding failures or any crud released from the fuel rods would remain in the pool and not be released as airborne releases. Particulates released into the pool will be controlled by the pool water treatment and cooling system as described in SAR Section 1.2.5.3.2. The worker dose assessments assume a dose rate of 2 mrem/hr while working on the spent fuel transfer machine to account for particulates in the water and direct radiation from assemblies in the pool.

For fuel cladding failures consisting of pinhole leaks or hairline cracks where the fission gases have not been fully released, failed fuel rods could release fission product gases into the WHF pool during spent fuel handling operations in the pool. However, for those small cladding failures during the short time period in the pool, the amounts of releases of fission product gases to the WHF would be insignificant. This is supported by the *Consolidated Safety Analysis Report for Morris Operation* (GE 2000, Section 7.3.3), which states that there had been no detectable leaking of gases from stored fuel in over 27 years of pool fuel storage. Event sequence analyses identify no events that would result in additional fuel cladding damage during normal assembly handling operations in the pool. Therefore, no significant additional exposure due to gaseous releases is expected during normal assembly handling operations in the pool.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

GE 2000. *Consolidated Safety Analysis Report for Morris Operation*. NEDO-21326. ADAMS ML080170476.

RAI Volume 2, Chapter 2.1.1.5, Second Set, Number 5:

Clarify the statement in SAR Section 1.8.3.1.3 (*Public Direct Radiation Methodology*) that the dose rates outside the facilities and within the GROA, were calculated using the computer code MCNP5. In BSC (2007am), DOE provides the calculated GROA dose rates and provides, in Table 9, a listing of TEDE exposures for the various onsite locations. These include the 78 mrem value at the lower muck yard and the 36 mrem at the switchyard. This table appears to be the source document for SAR Table 1.8-32. According to BSC (2007am, Section 4.2.1), the dose rates in Table 9 were calculated using the SCALE 4.4A computer code.

1. RESPONSE

Direct radiation dose rates outside facilities and within the geologic repository operations area are determined in *GROA External Dose Rate Calculation* (BSC 2007). Table 2 of the dose rate calculation (BSC 2007) provides dose rates versus distance for pads of the Aging Facility containing arrays of Transnuclear TN-32 dry storage casks, and Table 3 of the dose rate calculation (BSC 2007) provides dose rates versus distance for a single Transnuclear TN-32 dry storage cask. The dose rates in Tables 2 and 3 are calculated using the MCNP computer code.

Attachment I of the dose rate calculation (BSC 2007) uses the SAS1 module of the SCALE 4.4A computer code system to calculate the dose rate at 100 m for a single representative aging overpack containing a transportation, aging, and disposal (TAD) canister. The direct radiation dose rates versus distance for the Aging Facility containing representative aging overpacks with TAD canisters are determined by adjusting the dose rates versus distance for an aging pad containing arrays of Transnuclear TN-32 dry storage casks (using MCNP) as shown in Equation 1. The adjustment is the ratio of the dose rate at 100 m for a TAD canister aging overpack (using SCALE 4.4A) to the dose rate at 100 m for a single Transnuclear TN-32 cask (using SCALE 4.4A). This process is described in detail in the dose rate calculation (BSC 2007, Sections 4.3.1 and 6.1.1). Thus, the dose rates in Table 9 of the dose rate calculation (BSC 2007) were determined using both SCALE 4.4A and MCNP computer codes.

$$DR_{AF}^{TAD}(r) = \frac{DR^{TAD}(100)}{DR^{TN32}(100)} \times DR_{AF}^{TN32}(r) \qquad \text{(Eq. 1)}$$

where

$DR_{AF}^{TAD}(r)$ = dose rate from the Aging Facility loaded with representative TAD canister aging overpacks

$DR^{TAD}(100)$ = dose rate at 100 m from a single TAD canister aging overpack

$DR^{TN32}(100)$ = dose rate at 100 m from a Transnuclear TN-32 cask

$DR_{AF}^{TN32}(r)$ = dose rate from the Aging Facility loaded with Transnuclear TN-32 casks

The dose rate calculation (BSC 2007, Section 4.3.2) also uses direct radiation dose rates for transportation casks in the railcar and truck buffer areas, which are calculated using the MCNP computer code.

In summary, the direct radiation dose rates in SAR Table 1.8-32 outside facilities and within the geologic repository operations area are calculated in the dose rate calculation (BSC 2007) using the MCNP computer code to determine dose rates versus distance. The SCALE 4.4A computer code is used to determine a normalization factor for the Aging Facility.

2. COMMITMENTS TO NRC

The DOE commits to revising SAR Section 1.8.3.1.3 in a future update of the license application as described in Section 3.

3. DESCRIPTION OF PROPOSED LA CHANGE

Correct the first sentence of the second paragraph of Section 1.8.3.1.3 to read:

> The SAS1 module of the SCALE 4.4A computer code system and the MCNP computer code (Briesmeister 1997) are used to calculate direct and skyshine neutron and gamma dose rates at distances from rectangular arrays of aging overpacks.

Correct “MCNP5” to “MCNP” in the first sentence of the third paragraph of Section 1.8.3.1.3.

4. REFERENCES

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

Briesmeister, J.F., ed. 1997. *MCNP—A General Monte Carlo N-Particle Transport Code*. LA-12625-M, Version 4B. Los Alamos, New Mexico: Los Alamos National Laboratory. ACC: MOL.19980624.0328.

RAI Volume 2, Chapter 2.1.1.6, Second Set, Number 4:

> Provide a technical basis for not considering the failure of the programmable logic controller (PLC) on the transport and emplacement vehicle (TEV), for a scenario in Node Item No. 1.6 for base plate not retracting (Table E-2 of BSC 2008b).
>
> In Table E-2 of BSC 2008b, DOE has considered a scenario, labeled as Node Item 1.6, for TEV operations, in which the base plate does not retract, causing the shield door to remain open. The scenario then assumes that the PLC interlock will preclude TEV movement, and potential exposure to workers. DOE, however, has not provided a technical basis for not considering the PLC failure.

1. RESPONSE

Hazard and operability (HAZOP) evaluations are an early step in a detailed event sequence analysis that involves further evaluation of HAZOP identified deviations. The Node Item Number 1.6 (BSC 2008, Table E-2) refers to the HAZOP node where a potential direct worker exposure event sequence was hypothesized if the base plate does not retract and the transport and emplacement vehicle (TEV) moves. The HAZOP team documented the potential initiating event as such, and identified a programmable logic controller (PLC) interlock as a potential design feature to prevent movement of the TEV with the base plate extended and the shield doors open. This PLC interlock is not included in the current TEV design since the TEV has operational requirements that it be able to move with the base plate extended and shield doors open to facilitate the loading and unloading of a waste package when inside a loadout room, or an emplacement drift, respectively. During loading, the waste package and emplacement pallet sit stationary on the loadout room floor and the TEV (with base plate extended and shield doors open) drives up to and partially encloses the waste package. The waste package is subsequently drawn into the TEV via a motorized transfer carriage. When unloading a waste package, the TEV lowers the waste package and emplacement package to the floor of the emplacement drift and reverses away from the waste package. The base plate is retracted and the shield doors are closed once the TEV has cleared the emplaced waste package and emplacement pallet. When the TEV is outside a loadout room or an emplacement drift, its shield doors cannot be opened and, consequently, the base plate cannot be extended by inadvertent operator action or spurious PLC signal due to the incorporation of a permissive switch which is mechanically actuated inside a loadout room or an emplacement drift (SAR Sections 1.2.4.2.4.2, 1.3.3.5.2.1, and 1.3.3.5.2.3).

Hence, for the potential initiating event hypothesized during the HAZOP, it was determined that potential worker exposure was only of concern during loading of the waste package into the TEV in the Initial Handling Facility (IHF) or Canister Receipt and Closure Facility (CRCF) loadout rooms, or during waste package unloading from the TEV in the emplacement drift. The event occurring during waste package loading does not result in direct worker exposure as stated in *Subsurface Operations Event Sequence Development Analysis* (BSC 2008, Table E-2, Node Item Number 1.6) because there are no workers present during loading of a waste package into the TEV since this transfer process is performed remotely inside the IHF or the CRCF. Inadvertent worker access to the loadout room during waste package loading is prevented by the following operational requirements and engineering design features:

(1) Personnel clearing of the loading area is verified prior to commencing loading (SAR Table 1.9-10, PSC-3, and SAR Sections 1.2.3.2.4.4 and 1.2.4.2.4.4).

(2) Important to safety (ITS) interlocks between loadout room personnel access and equipment shield doors, and radiation monitors prevent opening of the loadout room shield doors during waste package loading (SAR Tables 1.4.2-1, 1.2.3-3, and 1.2.4-4 and SAR Figures 1.2.4-83, 1.2.4-86, and 1.2.4-87).

Similarly, the event occurring during emplacement does not result in direct worker exposure because workers are not present during the emplacement of the waste package since the TEV is controlled remotely and the turnouts and emplacement drifts are restricted access areas.

In the event that the TEV is moved after the waste package is loaded with the base plate extended and the TEV shield doors open, the ITS interlocks between the loadout room shield doors and the loadout room radiation monitors prevent opening of the loadout room shield doors, and subsequent worker access into, or departure of the TEV from, the loadout room. The loadout room equipment shield doors are also designed not to dislodge due to an impact from a fully loaded TEV travelling at maximum speed (1.7 mph). The PLC only controls the movement of the TEV and does not affect the ITS interlocks with the loadout room shield doors. Therefore, a failure of, or spurious signal from, the PLC would not result in the TEV moving outside of the loadout room with the base plate extended and the TEV shield doors open.

Failure of the TEV PLC resulting in inadvertent movement of the TEV with the base plate extended and the shield doors open has been considered in the preclosure safety analysis, and it has been determined that the design of the TEV, IHF, CRCF, and emplacement drifts adequately protects workers from inadvertent exposure to high radiation. The PLC discussed in Node Item 1.6 (BSC 2008, Table E-2) is not relied upon to preclude TEV movement with the base plate extended. Therefore, the TEV PLC is not relied upon to reduce the frequency of occurrence or mitigate the consequences of an event sequence, and consequently the TEV PLC is non-ITS.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

BSC (Bechtel SAIC Company) 2008. *Subsurface Operations Event Sequence Development Analysis*. 000-PSA-MGR0-00400-000-00A. Las Vegas, Nevada: Bechtel SAIC Company. ACC: ENG.20080214.0004.