9.0 RADIATION SAFETY

9.1 CONDUCT OF REVIEW

This chapter of the revised draft Safety Evaluation Report (DSER) contains the staff's review of radiation safety features described by the applicant in Chapter 9 of the revised Construction Authorization Request (CAR) (Reference 9.3.22). The objective of this review is to determine whether the applicant's radiation safety design features for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF or the facility) and radiation protection program is adequate to protect the radiological health and safety of workers, and whether the principle structures, systems, and components (PSSCs) and their design bases identified by the applicant provide reasonable assurance of worker protection against natural phenomena and the consequences of potential accidents.

As previously discussed in revised DSER Chapter 1 (see Section 1.1.1.1.1), the set of Nuclear Regulatory Commission (NRC) radiation safety requirements applicable to an individual depends on whether that individual is a worker exposed to radiation as part of his assigned employment duties. The radiological safety of such workers is the subject of this revised DSER Chapter. Further, for the purposes of this revised DSER, the staff has adopted the applicant's usage for names of individuals for whom accident risks must be limited. That is, "facility workers," are workers in the restricted area. "Site workers" are either: (1) members of the public in the controlled area who have received 10 CFR 70.61(f)(2) training and for whom the 10 CFR Part 19 posting requirements are met; or (2) workers in the controlled area. Site workers are treated as workers for the purposes of applying the NRC performance requirements of 10 CFR 70.61. "Public" or "members of the public" refers to individuals at the controlled area boundary. A fourth receptor protected under the provisions of 10 CFR 70.61 is the environment, which is the area outside the restricted area. The staff's evaluation of radiation safety protection of the public and the environment is provided in Chapter 10 of this revised DSER.

The staff evaluated the information provided by the applicant for radiation safety by reviewing Chapter 9 of the revised CAR, other sections of the revised CAR, supplementary information provided by the applicant, and relevant documents available at the applicant's offices but not submitted by the applicant. The review of radiation safety design bases and strategies was closely coordinated with the review of accident sequences described in the Safety Assessment of the Design Basis (see Chapter 5.0 of this revised DSER).

The staff reviewed how the radiation safety information in the revised CAR addresses or relates to the following regulations:

- Section 70.23(b) of 10 CFR states, as a prerequisite to construction approval, that the design bases of the PSSCs and the quality assurance program be found to provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

- Pursuant to 10 CFR 70.61(b)(1), the risk of credible high-consequence events to workers must be limited. For workers, a high consequence event is an internally or externally initiated event that results in an acute 100 rem total effective dose equivalent (TEDE). Controls must be used which either make the occurrence of such events highly unlikely, or make their consequences less severe than an acute 100 rem TEDE.

- Pursuant to 10 CFR 70.61(c)(1), the risk of credible intermediate-consequence events to workers must be limited. For workers, an intermediate-consequence event is an internally or externally initiated event that is not a high consequence event that results in an acute 25 rem TEDE . Controls must be used which either make the occurrence of such events unlikely, or make their consequences less severe than an acute 25 rem TEDE.

The review of this revised CAR focused on the design basis of radiation safety systems, components, and other related information. For radiation safety systems, the staff reviewed information provided by the applicant for the safety function, system description, and safety analysis. The review also encompassed proposed design basis considerations such as redundancy, independence, reliability, and quality. The staff used Chapter 9 in NUREG-1718 as guidance in performing the review (see Reference 9.3.10).

9.1.1 Radiation Safety Design Features

The objective of the staff's review under this section was to determine whether the applicant's design for construction and operation of the proposed facility is adequate to protect the radiological health and safety of workers (including site workers) from accidents and credible natural phenomena hazards. Specifically, the staff's review of the applicant's proposed radiation safety design features of the facility focused on PSSCs required to reduce the risk from intermediate consequence events (those producing an acute 25 rem TEDE to a worker) and high consequence events (those producing an acute 100 rem TEDE to a worker).

9.1.1.1 ALARA Design Considerations

The applicant committed to designing the MFFF in accordance with as low as reasonably achievable (ALARA) considerations. Below, the staff discusses the applicant's commitment to ALARA design considerations that might affect the design of radiation safety PSSCs.

The staff evaluated organizational relationships and responsibilities with respect to performing reviews of radiation safety design features, application of ALARA into design-stage collective dose estimates, descriptions and elements of the design review process for radiation safety design features and how the applicant proposes to use experience from past designs and from operating plants to develop improved radiation safety design features.

The applicant identified the organizational relationships and responsibilities for performing radiological design and design reviews in Section 9.1.1.1 of the revised CAR. These relationships and responsibilities are consistent with those described in the previously approved MOX Project Quality Assurance Plan (MPQAP), Revision 3 (see Reference 9.3.5, Section 1.0, and Reference 9.3.19, Section 1), and are thus acceptable.

The applicant's approach to design-stage collective dose estimates includes performing dose assessments in accordance with NRC Regulatory Guides 8.19 and 8.34 (see References 9.3.16 and 9.3.17), using the ABAQUES method and radiation zoning criteria for minimizing the direct (external) component of collective radiation dose, and using reviews of experience at the MELOX facility and considerations of facility design to evaluate and minimize collective internal doses.

The staff evaluated the applicant's approach to design-stage collective dose estimates. The staff finds this approach is acceptable because it will reduce the time spent in radiation areas, improve accessibility to components requiring periodic maintenance or inservice inspection, reduce the retention of radioactivity throughout plant systems, reduce contamination, facilitate decommissioning and minimize secondary waste production, and incorporate experience from operating plants and past designs. This approach is consistent with the acceptance criteria in Section 9.1.4.1.3 of NUREG-1718 (Reference 9.3.10).

With respect to external doses, the applicant did not identify sources of direct radiation that pose an unacceptable accident or natural phenomena risk. Therefore, the ABAQUES method and radiation zoning criteria are not design bases for PSSCs identified in the safety assessment. The applicant's ALARA design philosophy to prevent or mitigate the hazard of radioactive material intake will rely primarily on experience at the MELOX mixed applicant fuel fabrication facility in Marcoule, France. General measures and design philosophies for reducing the risk posed by a loss of material confinement that are described in the revised CAR include remote operations and automation, use of gloveboxes with negative pressure relative to occupied spaces, preventative maintenance, and airborne radioactivity monitoring. Staff has reviewed and evaluated the sources of radioactive material that the applicant proposes to use, namely weapon grade plutonium and depleted uranium. Staff agree that direct radiation from these sources will not pose an unacceptable accident or natural phenomena risk.

9.1.1.2 Facility Design Features

The information required to support the staff's review of the applicant's safety assessment includes facility and process drawings and descriptions with radiation safety design features that are PSSCs relied upon in the safety assessment. In the revised CAR, the applicant provided facility and process drawings and descriptions for the locations and access control points for restricted areas, the controlled area, passive and active material confinement boundaries, including ventilation system drawings. This information facilitated a clear understanding of the intended function of radiation safety PSSCs, commensurate with the expected level of design detail available at the construction authorization stage. The level of information provided by the applicant is, therefore, acceptable to the staff.

9.1.1.3 Source Identification

The applicant proposes that the plutonium applicant that would be shipped to an operating facility would contain mostly plutonium isotopes (Pu-238, Pu-239, Pu-240, etc.) and small quantities of other impurities. The impurities will include elements and radionuclides such as gallium, and isotopes of uranium and americium and radioactive decay products of uranium, plutonium and americium. The mass distribution of the radionuclides used by the applicant in their safety assessment is provided in revised DSER Table 9.1-1 (Reference 9.3.22, Table 9-3).

The staff confirmed by calculation that the radioactive decay progeny of the plutonium the applicant proposes to use would not make a significant contribution to the dose to workers resulting from accidents at the proposed facility. The staff calculated that the radionuclides listed in revised DSER Table 9.1-1 would account for greater than 99.99 percent of the dose resulting from inhalation. Therefore, for the purposes of the safety assessment, the staff considers the applicant's proposed list of radionuclides and mass fractions to be complete.

The aqueous polishing (AP) process would remove the impurities from the plutonium applicant as the first step in the fuel fabrication process. The "purified" form of plutonium applicant would be transferred to the powder blending and fuel fabrication process. The impurities would be temporarily stored in tanks in solution form before being transferred to the proposed Waste Solidification Building (WSB) for further treatment.

Depleted uranium (DU), which is < 0.7 percent by mass uranium-235, would be used in the fuel fabrication process. Depleted uranium nitrate solution [$DU(NO_3)_2$] would be used to dilute high-enriched uranium that would be separated from plutonium applicant feedstock in the aqueous polishing process. All tanks containing DU nitrate would be in the AP process.

DU is also the major component of fuel that would be manufactured at the facility. DU dioxide (DUO_2) powder would be used directly in the powder blend and pellet manufacturing process. The powder would comprise at least 94 percent of the mass of each fuel pellet. Given a MFFF design

Table 9.1-1.
Mass Fractions of Isotopes in Aged Weapons Grade Plutonium (WGPu)

Isotope	0 yr old WGPu, gm / gm WGPu	40 yr old WGPu, gm / gm WGPu	70 yr old WGPu, gm / gm WGPu
Pu-236	1.00E-09	5.98E-14	4.07E-17
Pu-238	6.86E-04	5.00E-04	3.95E-04
Pu-239	9.21E-01	9.20E-01	9.20E-01
Pu-240	6.18E-02	6.15E-02	6.14E-02
Pu-241	1.00E-02	1.00E-02 (1)	1.00E-02 (1)
Pu-242	1.00E-03	1.00E-03	1.00E-03
U-235	0.00E+00	1.83E-02 (2)	1.90E-02 (2)
U-238	0.00E+00	1.26E-03 (2)	1.26E-03 (2)
Am-241	0.00E+00	7.00E-03	7.00E-03 (1)

1. The applicant has made a conservative assumption that Pu-241, which has a 14.4 year half-life, does not decay.

2. The mass fraction of uranium isotopes represents ingrowth from the radioactive decay of plutonium-239 and plutonium-242 and isotopes that would be present in MFFF feedstock for other reasons.

Given a facility design production rate of 70 metric tons heavy metal (MTHM) of fuel per year, this corresponds to an annual demand for 65.8 tons of DU dioxide feedstock.

***Text removed under 10 CFR 2.390.**

In summary, the applicant identified six categories of radioactive sources at the facility that would be present in quantities that could pose unacceptable risks to workers from accidents and natural phenomena hazards (NPHs). These are the impure plutonium applicant [PuO_2(U)], the purified plutonium applicant [PuO_2(P)], the AP high alpha activity waste stream containing uranium, americium and other radioactive decay products [HAAW] depleted uranium applicant, the master blend and the final blend. The purified plutonium applicant source category, master blend and final blend represent three distinct concentrations of plutonium applicant (in decreasing concentration of the plutonium applicant): (1) a 100 percent plutonium applicant powder; (2) a 20 percent plutonium applicant-depleted uranium applicant (MOX) mixture; and

(3) a 6 percent MOX final blend. The isotopic mass distribution of radionuclides in each source category is provided in revised DSER Table 9.1-2.

The applicant provided an estimate of the amount of licensed material from each source category that could be present in each processing unit throughout an operating facility. For units in which multiple sources may be present, only the source likely to result in the bounding accident consequences was identified.

Table 9.1-2.
Mass Fractions of Isotopes in Each Source Category for the Safety Assessment

Radio-nuclide	PuO_2(U) (1)	PuO_2(P) (2)	High Alpha Activity Waste	Depleted Uranium (3)	20% MOX	6% MOX
Pu-236	4.07E-17	4.10E-17	-----	-----	8.20E-18	2.46E-18
Pu-238	3.95E-04	3.98E-04	-----	-----	7.96E-05	2.39E-05
Pu-239	9.20E-01	9.27E-01	-----	-----	1.85E-01	5.56E-02
Pu-240	6.14E-02	6.18E-02	-----	-----	1.24E-02	3.71E-03
Pu-241	1.00E-02	1.01E-02	-----	-----	2.02E-03	6.06E-04
Pu-242	1.00E-03	1.00E-03	-----	-----	2.00E-04	6.00E-05
U-235	(4)	-----		2.500E-03	2.00E-03	2.35E-03
U-238	(4)	-----		9.975E-01	7.98E-01	9.38E-01
Am-241	7.00E-03	-----	0.78 g/L	-----	-----	-----

1. PuO_2(U) is unpolished plutonium applicant feed
2. PuO_2(P) is polished plutonium applicant
3. Depleted uranium is both a feed material and a waste stream. These values are for feed.
4. Uranium isotopes do not contribute significantly to dose from inhalation events as compared to isotopes of plutonium and americium

The applicant also provided the photon and neutron energy spectra and intensities per unit mass of licensed material associated with both purified and impure plutonium applicant (revised DSER Tables 9.1-3 and 9.1-4). The photon energy spectra are based on the decay of the isotopes present in freshly-separated weapons-grade plutonium that would occur over a 70-year period. This accounts for the ingrowth of beta and gamma-emitting radionuclides that would contribute to direct gamma radiation exposures. The applicant also included in these spectra the contribution from (α,n) interactions in the applicant form. The staff independently verified the intensity and energy distribution of these spectra using simple spreadsheet models and the mass distributions of isotopes provided by the applicant. The staff finds that these spectra provide an acceptable basis for the proposed shielding design.

Energy spectra and source intensities per unit mass of depleted uranium were not provided by the applicant. However, given the low specific activity of this material, the staff does not anticipate that depleted uranium compounds would pose a direct radiation hazard to workers that would challenge the 10 CFR Part 70 performance requirements.

Another consideration in the hazard evaluation of radioactive sources is the solubility of inhaled radioactive compounds. In general, inhalation of soluble compounds of plutonium results in a committed dose approximately twice as high as the inhalation of insoluble compounds, namely plutonium applicant. However, soluble plutonium compounds, such as plutonium nitrate, tend to have higher molecular weights, which results in lower radioactivity per gram molecular weight of compound. To simplify the safety assessment, the applicant demonstrated a conservative assumption that the licensed nuclear material in all process units of the proposed facility would be insoluble plutonium applicant.

Table 9.1-3
Photon Energy Spectrum for MFFF Mixed Oxides

Photon Energy (MeV)	Intensity (gamma/sec/gram)	
	20% MOX	5% MOX
0.015	4.79E+07	1.20E+07
0.025	7.03E+01	1.76E+01
0.038	4.94E+04	1.23E+04
0.058	1.27E+05	3.17E+04
0.085	1.36E+04	3.40E+03
0.125	4.21E+04	1.06E+04
0.225	5.93E+03	1.57E+03
0.375	2.72E+04	6.80E+03
0.575	1.02E+03	2.55E+02
0.85	1.39E+02	3.47E+01
1.25	1.25E+01	3.13E+00
1.75	5.67E+00	1.42E+00
2.25	3.24E+00	8.14E-01
2.75	1.86E+00	4.67E-01
3.5	1.65E+00	4.13E-01
5	6.94E-01	1.74E-01
7	7.84E-02	1.97E-02
9.5	8.91E-03	2.24E-03
Total	4.82E+07	1.20E+07

Table 9.1-4
Neutron Energy Distribution for MFFF Mixed Oxides

Neutron Source	Intensity (neutrons/sec/gram)	
	20% MOX	5% MOX
Spontaneous Fission	11.8	2.96
(α, n)	12.0	3.0
Total	23.8	5.96

The applicant proposed the following relation:

$$[(\text{specific activity})\text{x}(DCF_Y)]_{\text{applicant}} \ \$ \ [(\text{specific activity})\text{x}(DCF_W)]_{\text{nitrate, oxalate, et al.}}$$

where the specific activity refers to the quantity of radioactivity per gram molecular weight of the compound and DCF_Y and DCF_W are the dose conversion factors for transportability class Y and class W, respectively, from Federal Guidance Report No. 11 (see Reference 9.3.4). This relation demonstrates that, for worker radiation dose calculations in the safety assessment, the applicant's assumption that all plutonium compounds are plutonium dioxide is conservative. This is because a mass unit of plutonium in the insoluble applicant form (class Y) results in a higher dose than a mass unit of plutonium in higher molecular weight soluble compounds (class W). The staff evaluated this assumption and find that it holds for the types of licensed nuclear materials proposed for use in the facility. Therefore, for the purposes of the safety assessment, the applicant's use of dose conversion factors for the least soluble compounds is acceptable to the staff.

DU compounds also pose an inhalation hazard to workers. However, the limiting hazard in the case of depleted uranium dioxide is chemical toxicity, not radiological dose. For this reason, the staff's evaluation of the applicant's safety assessment for this and other hazardous chemicals is provided in Chapter 8.0 of this revised DSER.

9.1.1.4 Safety Assessment of the Design Bases for PSSCs

The methodology presented by the applicant for their safety assessment of the design bases is consistent with the methodology presented in the Nuclear Fuel Cycle Facility Accident Analysis Handbook (see Reference 9.3.12). The following sections describe the staff's evaluation of the applicant's methodology for radiological consequence assessment in support of the safety assessment of the design bases.

With the exception of the postulated criticality accident, which involves a significant direct radiation hazard to facility workers, the radiation hazard of most concern in the safety assessment is the release of airborne radioactive material to occupied spaces. Therefore, the staff focused their review on the applicant's estimates of potential accidental releases of radioactive material from engineered confinement systems.

9.1.1.4.1 Facility Worker Consequence Assessment

To reduce the risk to the facility worker, the applicant has proposed a methodology in the safety assessment, described in Section 5.4.4.2 of the revised CAR, in which event consequences to facility workers are qualitatively estimated. Since the applicant assumes the facility worker would be at the location of the release, the risks from events involving plutonium and americium are deemed to be qualitatively unacceptable and PSSCs are deterministically applied. Further, the applicant assumed that unmitigated consequences from the release of uranium are low and PSSCs were not applied.

However, the applicant's safety assessment includes several events in which the proposed PSSC for protection of the facility worker is Facility Worker Action to mitigate the consequences. For example, in the event of a loss of radioactive material confinement involving a fuel assembly drop, waste container drop, or a small loss of confinement at a glovebox, the worker is expected to recognize the event and leave the affected area. For these events, the staff requested quantitative dose consequence estimates in order to evaluate the adequacy of this safety strategy. The staff's evaluation of this approach is discussed further in revised DSER Section 9.1.2.3, "Training."

9.1.1.4.2 Accident Release Estimates for Site Worker Radiation Safety Features

A five-factor approach was used to estimate accidental releases of radioactivity to the atmosphere which would result in a site worker immediately outside the restricted area receiving a dose. The total quantity release or the release rate over time is referred to as the source term.

The source terms derived in the applicant's assessment is the product of five parameters. The first parameter is the quantity of radioactive material that is postulated to be involved in the event. The remaining four parameters are reduction factors that are applied to account for different physical phenomena that would limit the amount of licensed nuclear material to which a facility or site worker would be exposed. The general form of the source term formula is:

Source Term, kilograms = MAR x DR x ARF x RF x LPF   Eq. 9-1

where MAR is the quantity of material at risk during the event, in kilograms,

DR is the unitless damage ratio of material actually impacted by the event (0<DR<1),

ARF is the atmospheric release fraction, which is the unitless fraction of impacted material (MAR x DR) that can become airborne (0< ARF<1),

RF is the respirable fraction, which is the unitless fraction of airborne material that can be inhaled into the human respiratory system (0<RF<1), and

LPF is the unitless fraction of airborne material that breaches the confinement barrier.

The material-at-risk (MAR) was provided by the applicant for over 200 individual process units throughout the proposed facility. These included the six major types of radioactive sources as described in revised DSER Section 9.1.1.3. The applicant's hazard analysis considered many different types of events that could cause an adverse human health or environmental effect as a result of accidental exposure to these radioactive sources. These types of accidents were categorized into the following major event categories: fires, explosions, loss of confinement, load drops, and nuclear criticality. The staff evaluation of the applicant's hazard assessment, including event types that were screened from further consideration, is provided in Chapter 5.0 of this revised DSER.

The damage ratio (DR) was set equal to one, with the following exceptions: the DRs assumed for pellets exposed to overpressurization gas flows and pressurized rods that have been breached are 0.01 and 0.001, respectively. If rods are dropped, the DR is assumed by the applicant to be equal to 0.02. The applicant also set the DR equal to 0.1 for a fire in the Secured Warehouse Building. The staff has verified that the applicant's assumptions are consistent with staff's guidance provided in NUREG/CR-6410 (see Reference 9.3.12). Therefore, these assumptions are acceptable to the staff.

The applicant chose values for the atmospheric release fraction (ARF) and respirable fraction (RF) based on values in Reference 9.3.12. The values chosen by the applicant for these parameters are provided in revised DSER Table 9.1-5. The staff reviewed the applicant's basis for choosing ARFs and RFs and found them to be consistent with Reference 9.3.12. The applicant selected a value of the ARF for dropped fuel rods to be 3.0×10^{-5}. The basis for this value is a study performed by Sandia National Laboratory (Reference 9.3.20). These assumptions are acceptable to the staff because they are based on either data previously used and accepted by the staff for safety assessments or on data derived from experiments related to the actual hazard.

Table 9.1-5
Atmospheric Release Fractions and Respirable Fractions

Release Form	Release Fraction	Explosive Detonation	Explosive Over-pressurization	Fire/Boil	Drop	Entrainment
Solution	ARF	1.0	5.0×10^{-5}	2×10^{-3}	2.0×10^{-5}	4.0×10^{-7}
	RF	0.01	0.8	1.0	1.0	1.0
	ARF x RF	0.01	4.0×10^{-5}	2×10^{-3}	2.0×10^{-5}	4.0×10^{-7}
Powder	ARF	1.0	5.0×10^{-3}	6.0×10^{-3}	2.0×10^{-3}	4.0×10^{-5}
	RF	0.2	0.3	0.1	0.3	1.0
	ARF x RF	0.2	1.5×10^{-3}	6.0×10^{-4}	6.0×10^{-4}	4.0×10^{-5}
Pellet	ARF	0.01	5.0×10^{-3}	5.0×10^{-4}	1.0	NA
	RF	1.0	0.3	0.5	1.1×10^{-5}	
	ARF x RF	0.01	1.5×10^{-3}	2.5×10^{-4}	1.1×10^{-5}	
Rod	ARF	0.01	3.0×10^{-5}	0.0	3.0×10^{-5}	NA
	RF	1.0	1.0	1.0	1.0	
	ARF x RF	0.01	3.0×10^{-5}	0.0	3.0×10^{-5}	
Filter (unencased)	ARF	2.0×10^{-6}	0.01	1.0×10^{-4}	1.0×10^{-2}	NA
	RF	1.0	1.0	1.0	1.0	
	ARF x RF	2.0×10^{-6}	0.01	1.0×10^{-4}	1.0×10^{-2}	

As used by the applicant, the leak path factor generally accounts for the particulate matter removal efficiency of the high-efficiency particulate air (HEPA) filters used in the ventilation confinement systems. When relied upon to mitigate the effects of an accident, the applicant assumed 99 percent removal efficiency (i.e., 1% leak path factor) per stage. Each HEPA system relied upon for safety includes two banks or stages of HEPA filters in series. Assuming that the effective leak path factor for a system of staged HEPA filters is the product of the individual leak path factors for successive filter stages, the applicant applied a leak path factor of 10^{-4} for systems relied upon in their safety assessment. The combination of efficiencies in this manner is acceptable to the staff, because it is consistent with the guidance in Reference 9.3.12, Section F.2.1.3, however, the staff has not accepted the value of 10^{-4}.

When prevention alone, rather than mitigation, was the applicant's preferred safety strategy, the applicant applied a leak path factor equal to zero. The applicant used a leak path factor equal to one when the HEPA filters were either unlikely to function as needed or not required to mitigate the event consequences (see "Verification of Low Consequence Events," revised DSER Section 9.1.1.4.4).

As described in Section 11.4 of this revised DSER, the staff has questioned the applicant's use of a 99 percent removal efficiency per stage during events that could challenge the function of the filters (Open Item VS-1). Appendix F of Reference 9.3.12 recommends an efficiency of between 99 percent and 95 percent for severe conditions. Therefore, for the purposes of this revised DSER, the staff analyzed the accident consequences for fire and explosion events using an leak path factor (LPF) of 0.01. The staff's evaluation of HEPA filter efficiencies is described in Section 11 of this revised DSER.

NRC Regulatory Guides 3.71 and 3.35 (see References 9.3.15 and 9.3.14) were used by the applicant to develop source terms for direct radiation and airborne releases resulting from a criticality accident. However, since NRC has withdrawn these guides, the staff used the current guidance in Reference 9.3.12 to estimate the downwind consequences to a site worker of a criticality accident. By so doing, the staff independently evaluated the applicant's source terms, and find that the applicant's analysis is consistent with the current guidance, and is therefore, acceptable.

9.1.1.4.3 Dose Assessment for the Site Worker

The applicant's methodology for dose assessment relies on an assumption that the principle human health hazard posed by releases of radioactive material from the proposed MOX facility is inhalation of radioactive material downwind of the facility. Other pathways of exposure would include direct radiation from the passing plume and exposure to ground surfaces contaminated by material depositing on the ground as the plume passes. However, the staff confirmed by calculation that, with the exception of the postulated criticality event, the direct radiation and ground contamination pathways are negligible as compared to the inhalation pathway.

To calculate the 50-year committed effective dose equivalent (CEDE) from inhalation doses from passing plumes, the applicant applied a simple formula involving the source term (Eq. 9.1), the atmospheric dispersion factor (χ/Q), a human receptor's breathing rate (B.R.), and the dose conversion factor (DCF) (from Reference 9.3.4):

$$CEDE_i\ [\text{rem}] = \text{Source Term}_i\ [\text{kg}] \times \chi/Q\ [\text{s m}^{-3}] \times \text{B.R.}\ [\text{m}^3\ \text{s}^{-1}] \times DCF_i\ [\text{rem}\ \mu\text{Ci}^{-1}] \times C_i\ [\mu\text{Ci kg}^{-1}]$$

where $CEDE_i$ is the committed dose from the ith radionuclide, and
C_i is the specific activity of the ith radionuclide.

Atmospheric dispersion factors were calculated by the applicant using site-specific meteorological data from the Savannah River Site (SRS) H-Area meteorological tower collected from 1987 through 1996. The ARCON96 model (see Reference 9.3.18) was used to estimate factors for the site worker located 100 meters from the plant stack. The value calculated by the applicant was 6.1E-4 s m^{-3}. The staff verified by independent calculations that the meteorological data used by the applicant in their safety assessment is consistent with data published by the U.S. Department of Energy's (DOE's) SRS for the H-Area meteorological tower (DOE, 1999). The staff also performed independent calculations for the site worker atmospheric dispersion factor and calculated a value of 6.1E-4 s m^{-3}. The staff used this value of the atmospheric dispersion factor to calculate the consequences from controlling events that are presented in Table 9.1-6 of this DSER.

The breathing rate of 3.47E-4 m^3 s^{-1} assumed by the applicant is consistent with guidance provided by the NRC in Regulatory Guide 1.25 (see Reference 9.3.13), and is equivalent to a volume of 10 cubic meters inhaled during an 8-hour workday. This assumption is based on NRC guidance applicable to fuel handling and is, therefore, acceptable to the staff for use in the applicant's safety assessment.

EPA dose conversion factors used by the applicant (Reference 9.3.4) are based on the recommendations of the International Commission on Radiation Protection (ICRP). These are the same recommendations that form the basis for NRC radiation protection standards in 10 CFR Part 20. Therefore, these factors are acceptable to the staff.

The source of values for C_i, the specific activity of the i^{th} radionuclide, were not provided by the applicant. The staff used information provided in ICRP Publication 38 (see Reference 9.3.8) in its independent evaluation.

The results of the staff's independent evaluation of bounding event consequences for site workers is provided in revised DSER Table 9.1-6. For many events, the PSSC applied to reduce the risk of the event would actually lower the likelihood of the event. A significant margin of safety exists for all of the mitigated events. The smallest margin is about a factor of ten between the 2.6 rem acute TEDE consequence to the site worker resulting from a fire and the 25 rem acute TEDE intermediate consequence threshold.

9.1.1.4.4 Verification of Low Consequence Events

Unmitigated event consequences result from an accident sequence when mitigative controls either fail or do not exist. Unmitigated event consequences are those consequences calculated by the applicant prior to determining and taking credit for PSSCs that would reduce the risk of the event. However, in some cases the unmitigated event consequence is so low that it falls below the intermediate consequence threshold values for workers specified in 10 CFR 70.61(c)(1). These events, referred to as "low" consequence events, require no PSSCs to lower the risk. The applicant identified 22 hazard assessment events as low consequence events. Sixteen of these were loss of confinement events, three were fires and three were load handling events.

The staff performed independent calculations to verify the applicant's assertion that some events would be low consequence events and would not require PSSCs to further reduce the accident risk. Based on the staff's confirmatory analysis, the staff accepts the applicant's categorization in its hazard assessment of the 22 events as being low consequence events.

9.1.2 Radiation Protection Program

The purpose of this review is to determine whether the applicant's radiation protection program is adequate to protect the radiological health and safety of the workers and to comply with the regulatory requirements of 10 CFR Parts 19, 20 and 70, to the extent such programmatic elements are relied upon to provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

The staff reviewed the revised CAR using the guidance in NUREG-1718, Section 9.2.

9.1.2.1 ALARA

The applicant's commitment to an ALARA program at the construction authorization stage includes a management commitment to this policy, an ALARA Committee, administrative control levels and dose limits for design, internal audits and assessments.

The staff has reviewed the applicant's brief description of the ALARA program and its related safety assessment of the design bases, in which the applicant did not identify any PSSCs or management measures within the purview of its ALARA program. The staff finds that, at the construction authorization stage, no such PSSCs or management measures are required. However, the regulations in 10 CFR Parts 20 and 70 contain specific requirements for an ALARA program that would have to be fully and adequately addressed in any subsequent application for a special nuclear material (SNM) possession and use license.

Table 9.1-6
Site Worker Bounding Consequences from Hazard Event Categories

Bounding Hazard Event	Site Worker, rem	
	Applicant	NRC
Loss of Confinement[a]	**	**
Internal Fire[b]	**	**
Load Handling	**	**
Criticality	**	**
Explosion[b]	**	**

[a] The bounding loss of confinement event is a load handling accident involving the Jar Storage and Handling Unit.

[b] NRC estimates of the values for the internal fire and explosion are approximately 100 times higher than the applicant's values. This is because staff estimated the filter efficiency of the final two HEPA filters to be 99% during these severe events.

**** Text removed under 10 CFR 2.390.**

9.1.2.2 Radiation Safety Procedures and Radiation Work Permits

The applicant identified seven major components of its radiation safety procedures and work control program in the revised CAR. These are work planning, entry and exit control, radiological work controls, posting and labeling, release of materials and equipment, sealed radioactive source accountability and control, and radioactive material receipt.

Of these controls, the entry and exit controls, specifically process cell entry controls, are relied on in the safety assessment of the design bases as a PSSC to protect the facility worker as shown in revised DSER Table 9.1-7. The applicant proposes several controlling parameters in the design basis for the Process Cell Entry Controls. To prevent access during normal operations, the applicant proposes Radiation Work Permits, signs and postings, and barricades. The reliance on a combination of administrative controls and engineered controls for this safety function provides adequate assurance that it will be highly unlikely for facility workers to inadvertently enter Process Cells upon failure of an administrative control. Therefore, this safety strategy is acceptable to the staff.

The applicant proposes to further control radiation dose during any maintenance activities at the proposed facility by committing to meet the requirements of 10 CFR 20.1601 and 10 CFR 20.1602 for high and very high radiation areas, respectively. The applicant's commitment to meet these regulatory requirements is acceptable to the staff.

9.1.2.3 Training

In the revised CAR, the applicant identified "Facility Worker Action" and "Facility Worker Controls" as PSSCs for several internal hazards at the proposed facility. As shown in Table 9.1-7, design basis controlling parameters for Facility Worker Action would include evacuation of the affected area. Facility Worker Controls are those actions taken by the facility worker prior to commencing an activity that could result in an event with unacceptable dose consequences. In any subsequent application for an SNM possession and use license, in accordance with 10 CFR 70.62(d), the applicant would be required to specify the management measures (such as training and procedures) which would adequately ensure that facility worker actions and controls are reliable to serve their intended safety function (i.e., the facility worker understands what to do and when to do it). The applicant clarified that where events are obvious to a facility worker, and the facility worker has adequate time to respond by taking self-protecting action, that action is credited in mitigating consequences to the facility worker. Examples of such events are assembly or rod handling events, waste drum handling and fire events, and glovebox fires. The staff reviewed the applicant's calculation of consequences to facility workers from these events, as described below.

The applicant's calculations, reviewed and confirmed by the staff, demonstrated low mitigated dose consequences to the facility worker for two events for which the staff requested additional information. These events are the postulated solid radioactive waste container drop and the drop of a completed fuel assembly. The solid waste used in this calculation is of the type likely to contain the most amount of plutonium compound, which is the compacted plutonium dioxide convenience cans.

The applicant's calculations demonstrate that the consequences would be adequately mitigated by the facility worker's response, which is to evacuate the affected area within 30 seconds. The applicant's calculation demonstrates an adequate margin of safety in that the dose to the facility worker would be at least ten times lower than the 25 rem intermediate consequence threshold for workers specified in 10 CFR 70.61(c)(1).

9.1.2.4 Air Sampling

The applicant has committed to air sampling for facility worker exposure monitoring and control. The applicant plans to use measurements of airborne radioactivity concentrations to estimate internal dose in accordance with 10 CFR 20.1204. However, the Airborne Radioactivity Monitoring Program was not identified as a PSSC by the applicant in their safety assessment.

The staff has reviewed the applicant's brief description of the Airborne Radioactivity Monitoring Program and its related safety assessment of the design bases, in which the applicant did not identify any PSSCs or management measures within the purview of this program. The staff finds that, at the construction authorization stage, no such PSSCs or management measures are required. However, the regulations in 10 CFR Parts 20 and 70 contain specific requirements for such a program that would have to be fully and adequately addressed in any subsequent application for an SNM possession and use license.

9.1.2.5 Contamination Control

The staff has reviewed the applicant's brief description of the Contamination Control Program and its related safety assessment of the design bases, in which the applicant did not identify PSSCs or management measures within the purview of this program. The staff finds that, at the construction authorization stage, no such PSSCs or management measures are required. However, the regulations in 10 CFR Parts 20 and 70 contain specific requirements for such a program that would have to be fully and adequately addressed in any subsequent application for an SNM possession and use license.

Table 9.1-7
Radiation Protection PSSCs

PSSC	Safety Function	Controlling Parameters
Process Cell Entry Controls (Administrative)	Prevent the entry of personnel into process cells during normal operations.	Radiation Work Permits Signs and Postings Barricades
	Ensure that facility workers do not receive a dose in excess of limits while performing maintenance in the AP process cells.	Radiation Work Permits Signs and Postings Commitment to 10 CFR 20.1602, for very high radiation areas
Facility Worker Action (Administrative)	Ensure that facility workers take proper actions to limit chemical and radiological exposure	Facility worker response to exit the affected area
Facility Worker Controls (Administrative)	Ensure that facility workers take proper actions prior to bag-out operations to limit radiological exposure. Ensure that facility workers take proper actions during maintenance activities to limit radiological exposure.	Facility worker pre-job preparation to prevent and/or limit dose during tasks involving transient primary confinements or maintenance in AP/MP C3 areas.

9.1.2.6 External Exposure, Internal Exposure, and Summing Internal and External Exposure

The applicant provided a brief description of their Direct Exposure Control Program, Internal Exposure Control Program, and procedures for summing dose contributions from both direct exposure and intake of radioactivity in the revised CAR. These programs are not identified as either PSSCs or management measures in their safety assessment.

Though not applicable to safety assessments for potential accidents, the applicant has provided estimates of total radiation doses to facility workers during normal operations based on experience during operation of a similar facility, the MELOX plant in Marcoule, France.

The applicant's estimates of external exposure hazards at the proposed MFFF involve an adjustment of the external dose rates at the MELOX plant to account for the different isotopic distribution of plutonium isotopes in reactor-grade plutonium used at MELOX, as compared to the weapons-grade plutonium proposed for use in the MFFF. This adjustment includes both a 20:1 MELOX:MFFF adjustment of the photon intensity and a 11:1 adjustment in the neutron intensity per unit mass of plutonium. With these adjustments, the external collective dose is estimated to be 12 person-rem per year, including a 10 person-rem contribution from the MOX processing units (MP) and 2 person-rem from the AP area.

For internal dose estimates, the applicant surveyed recent records at the MELOX mixed oxide fuel fabrication facility in Marcoule, France. From 1996 through July 2001, 41 individuals received an internal dose. Thirty workers received less than 10 percent of the allowable limit on intake (ALI) (based on a 2 rem annual limit (Reference 9.3.9), ten workers received between 10 percent and 33.3 percent of the ALI, and one worker received between 33.3 percent and 100 percent of the ALI. Using this data, the staff calculated a collective dose of approximately 13 person-rem. Given the five year period over which these intakes occurred, the average annual internal dose is approximately three person-rem for workers at the MELOX facility. The applicant estimates a value of 4.5 person-rem per year. The MELOX facility is larger than the proposed MFFF, and has a higher throughput of material. In addition, MELOX processes reactor grade plutonium, which delivers about ten times more dose from inhalation than the weapons grade plutonium proposed for use in the MFFF. Table 9.1-8 presents a comparison of the inhalation doses per unit mass of plutonium dioxide for reactor grade and weapon grade plutonium. Therefore, the staff concurs that this collective dose is a reasonable upper bound of doses expected at the MFFF.

Table 9.1-8
Comparison of the Radiotoxicity of Reactor Grade (RG) Plutonium and Weapon Grade (WG) Plutonium; Total Effective Dose Equivalent per Gram of Plutonium Inhaled

Pu Isotope	Dose Conversion Factor [rem/ μCi]	S.A. of Pu isotope [μCi/g]	RG Isotopic Mass Fraction	RG Dose Fraction [rem / g Pu]	WG Isotopic Mass Fraction	WG Dose Fraction [rem / g Pu]
Pu-238	$4x10^{-3}$	$1.71x10^{7}$	0.0149	$1.0x10^{3}$	0.000395	$2.7x10^{1}$
Pu-239	$4x10^{-3}$	$6.13x10^{4}$	0.595	$1.5x10^{2}$	0.92	$2.3x10^{2}$
Pu-240	$4x10^{-3}$	$2.28x10^{5}$	0.240	$2.2x10^{2}$	0.0614	$5.6x10^{1}$
Pu-241	$2x10^{-1}$	$1.03x10^{8}$	0.103	$2.1x10^{6}$	0.01	$2.1x10^{5}$
Pu-242	$4x10^{-3}$	$3.93x10^{3}$	0.04	$6.3x10^{-1}$	0.001	$1.6x10^{-2}$
TOTAL				**$2.1x10^{6}$**		**$2.1x10^{5}$**

Therefore, the applicant's estimate that the TEDE among facility workers at the proposed MFFF would be less than 20 person-rem is reasonable to the staff. The applicant further assumes the total workforce will number approximately 400 facility workers. Therefore, the average TEDE per facility worker is less than 50 mrem per facility worker per year, well below the occupational dose limits of 5,000 mrem set forth in 10 CFR 20.1201(a).

9.1.2.7 Respiratory Protection

As discussed above in Section 9.1.2.4, “Training,” the applicant proposes that facility worker actions be relied upon to mitigate the consequences of some facility hazards (Reference 9.3.7, page 16). The applicant did not explicitly identify design basis controlled parameters and values associated with facility worker respiratory protection equipment for this purpose. Rather, the applicant stated that facility workers are assumed to be in the impacted area for no more than 30 seconds and that they do not use their personnel protection equipment. The staff has reviewed the applicant’s calculations and agree that “Facility Worker Action,” an administrative PSSC that requires facility workers to escape the affected area, provides an adequate level of protection for the facility worker for those events where this PSSC is applied.

9.1.2.8 Instrumentation

The applicant has committed to an Instrumentation Calibration and Maintenance Program to ensure that radiation protection instrumentation is calibrated and maintained in accordance with American National Standards Institute (ANSI) 323.

The staff has reviewed the applicant's brief description of the Instrumentation Calibration and Maintenance Program and its related safety assessment of the design bases, in which the applicant did not identify PSSCs or management measures within the purview of this program. The staff finds that, at the construction authorization stage, no such PSSCs or management measures are required. However, the regulations in 10 CFR Parts 20 and 70 contain specific requirements for such a program that would have to be fully and adequately addressed in any subsequent application for an SNM possession and use license.

9.1.2.9 Issues Pertaining to NRC Radiation Safety Regulation at a DOE Site

The proposed facility would be owned by the DOE and built on the Savannah River Site, a 300 square mile reservation on which DOE already owns and operates a number of nuclear and non-nuclear facilities. Radiation safety at the other facilities on the SRS are regulated by the DOE in accordance with the regulations in 10 CFR Part 835 (Reference 9.3.2). However, in accordance with the 10 CFR Part 20 and 10 CFR Part 70, the applicant has designated a controlled area boundary for the facility on the SRS. The proposed controlled area boundary is largely contiguous with the SRS boundary. Therefore, individuals on the SRS will simultaneously occupy areas controlled by both the applicant and DOE. For this reason, these individuals could be subject to dual regulation under both 10 CFR Part 20 and the DOE’s regulations at 10 CFR Part 835.

For example, one issue that was addressed by the staff during the review of the revised CAR is the question of which radiation dose limit should be applicable to individuals located on the 300 square mile facility controlled area, most of whom work for the DOE on the SRS. As discussed above and in revised DSER Chapter 1, the NRC regulations governing dose limits make a distinction between workers and members of the public, based on a person’s assigned employment duties, rather than a person’s geographic location relative to the licensed facility. Therefore, in accordance with 10 CFR 20.1301(b), SRS employees within the facility controlled area whose assigned duties do not involve exposure to radiation or radioactive materials would be regarded by the NRC as members of the public. As such, these individuals would be subject to the dose limit of 100 mrem per year TEDE applicable to members of the public, as specified in 10 CFR 20.1301(a), rather than the higher occupational dose limits of 5,000 mrem per year specified in 10 CFR 20.1201(a).

In the revised CAR Section 1.1.2.1, the applicant stated that "both an SRS worker outside the MFFF but within the controlled area boundary, as well as an MFFF worker within the controlled area boundary, are deemed to be 'workers.'" Further, the applicant commits to "establish a protocol with the U.S. Department of Energy Savannah River Operations Office (DOE-SR) to ensure the following:

- "An augmentation of the existing SRS radiation protection training program to address MFFF-specific radiation risks in accordance with the provisions for instructions to workers of 10 CFR 19.12(a)(1)-(5). This specific training will ensure the awareness of SRS workers to the risks associated with accidents involving the MFFF licensed activities as determined by the Integrated Safety Analysis (ISA).

- "The posting and maintenance of notices in conspicuous locations within F area, in accordance with the posting provisions of 10 CFR §19.11(a)."

The staff accepts this commitment for meeting the 10 CFR 70.61 (f) requirement, conditional on the fact that NRC regards individuals, including SRS employees, whose assigned duties do not involve exposure to radiation or radioactive materials as members of the public. These individuals are also "individuals who are not workers," as the term is used in 10 CFR 70.61(f), and should also be subject to the training identified in the first bullet above.

The applicant will be required to more fully address its commitments to meet 10 CFR Part 20 requirements in any subsequent application it may file for an SNM possession and use license. Specifically, the applicant will be required to show that it has control in the controlled area sufficient to meet the requirements in 10 CFR Part 20 pursuant to an agreement with DOE, as proposed in the revised CAR.

A summary comparison of NRC and DOE radiation safety rules and regulations is provided in the staff's summary report on NRC involvement with DOE in the Hanford Tank Waste Remediation System Privatization (TWRS-P) program (Reference 9.3.11).

9.1.3 Radiation Safety Design Bases

The PSSCs for facility worker radiological protection which are elements of the applicant's proposed radiation protection program and that were identified in the applicant's safety assessment of the design bases are listed in revised DSER Table 9.1-7. This table lists the radiation safety PSSCs, functions and controlled parameters for design described in the revised CAR. The staff has reviewed whether these functions and parameters provide reasonable assurance that the applicant's design for construction and operation of the facility would be adequate to protect the radiological health and safety of workers and to comply with the regulatory requirements of 10 CFR Parts 20 and 70 during routine and nonroutine operations, including anticipated events. The staff's evaluation findings are provided in revised DSER Section 9.2.

9.2 EVALUATION FINDINGS

In Chapter 5 of the revised CAR, DCS provided design basis information for radiation protection that it identified as PSSCs for the proposed facility. Based on the staff's review of the revised CAR and supporting information provided by the applicant relevant to radiation protection, the staff finds that the design bases of the PSSCs identified by the applicant will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents, in accordance with 10 CFR 70.23(b).

As discussed above in revised DSER Section 9.1, pursuant to 10 CFR 70.61(b)(1) and 10 CFR 70.61(c)(1), the risk of credible high-consequence and intermediate-consequence events must be limited. For workers, a high consequence event is an internally or externally initiated event that results in an acute 100 rem TEDE, and an intermediate-consequence event is one which results in an acute 25 rem TEDE. For such high-consequence events, controls must be used which either make the occurrence of these events highly unlikely, or make their consequences less severe than an acute 100 rem TEDE. For such intermediate-consequence events, controls must be used which either make the occurrence of these events unlikely, or make their consequences less severe than an acute 25 rem TEDE. Based on staff's review of the safety assessment methodology presented in the revised CAR, the staff finds that the design bases of the PSSCs identified by the applicant will reduce facility worker dose consequences to acceptable levels, for both high-consequence and intermediate-consequence events. Accordingly, the staff concludes that for worker doses, the applicant has met the 10 CFR 70.61(b)(1) and 10 CFR 70.61(c)(1) performance requirements.

The following open item in the April 2002 DSER has been closed and is discussed in Appendix B: RS-1

9.3 REFERENCES

9.3.1 Code of Federal Regulations, *Title 10, Energy,* Part 20, "Standards for Protection Against Radiation."

9.3.2 _____. *Title 10 Energy,* Part 835, "Occupational Radiation Safety."

9.3.3 Department of Energy (U.S.) (DOE). DOE-STD-3009-2000, "DOE Standard: Stabilization, Packaging, and Storage of Plutonium-Bearing Materials." DOE: Washington, D.C. September 2000.

9.3.4 Environmental Protection Agency (U.S.) (EPA). EPA-520/1-88-020, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion." EPA: Washington, D.C. September 1988.

9.3.5 Hastings, P. S., Duke Cogema Stone & Webster, Letter to Nuclear Regulatory Commission, RE MOX Project Quality Assurance Program Plan, March 26, 2002.

9.3.6 Hastings, P. S., Duke Cogema Stone & Webster, Letter to Nuclear Regulatory Commission, RE DCS-NRC-000081, Clarification of Responses to NRC Request for Additional Information, January 7, 2002.

9.3.7 Hastings, P. S., Duke Cogema Stone & Webster, Letter to Nuclear Regulatory Commission, RE DCS-NRC-000083, Clarification of Responses to NRC Request for Additional Information, February 11, 2002.

9.3.8 International Commission on Radiation Protection (ICRP). ICRP Publication 38, "Radionuclide Transformations: Energy and Intensity of Emissions." ICRP:1983.

9.3.9 International Commission on Radiation Protection (ICRP). ICRP Publication 60, "1990 Recommendations of the International Commission on Radiological Protection." ICRP: 1991.

9.3.10 Nuclear Regulatory Commission (U.S.)(NRC). NUREG-1718, “Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility.” NRC: Washington, D.C. 2000.

9.3.11 _____. NUREG-1747, “Overview and Summary of NRC Involvement with DOE in the Tank Waste Remediation System-Privatization (TWRS-P) Program.” NRC: Washington, D.C. August 2001.

9.3.12 _____. NUREG/CR-6410, “ Nuclear Fuel Cycle Facility Accident Analysis Handbook.” NRC: Washington D.C. March 1998.

9.3.13 _____ Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)." NRC: Washington, D.C. 1972.

9.3.14 _____ Regulatory Guide 3.35, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Plutonium Processing and Fuel Fabrication Plant," NRC: Washington, D.C. 1972.

9.3.15 _____ Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," NRC: Washington, D.C. 1972.

9.3.16 _____. Regulatory Guide 8.19, “Occupational Radiation Dose Assessment in Light Water Reactor Power Plants Design Stage Man-Rem Estimates.” NRC: Washington, D.C. 1979

9.3.17 _____. Regulatory Guide 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses.” NRC: Washington, D.C. 1992.

9.3.18 _____. “ARCON96 Code System to Calculate Atmospheric Relative Concentrations in Building Wakes. CCC-664 Radiation Safety Information Computational Center.” NRC: Washington, D.C. 1999.

9.3.19 Persinko, A., Nuclear Regulatory Commission, Letter P. Hastings, Duke Cogema Stone & Webster, RE Safety Evaluation Report: Quality Assurance Program for Construction of MOX Fuel Fabrication Facility, January 10, 2003.

9.3.20 Sandia National Laboratories (SNL) SAND87-7082, “Effects of a Potential Drop of a Shipping Cask, a Waste Container, and a Bare Fuel Assembly During Waste-Handling Operations; Yucca Mountain Site Characterization Project.” SNL: December 1991.

9.3.21 Hastings, P., Duke Cogema Stone & Webster, letter to U.S. Nuclear Regulatory Commission, RE Clarification of Responses to NRC Request for Additional Information, April 23, 2002.

9.3.22 Ihde, R. Duke, Cogema Stone & Webster, letter to W. Kane, U.S. Nuclear Regulatory Commission, RE MOX Fuel fabrication Facility- Construction Authorization Request, October 31, 2002.

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