UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 12, 2004

MEMORANDUM TO: Martin J. Virgilio, Director
Office of Nuclear Material Safety
and Safeguards

FROM: Robert C. Pierson, Director
Division of Fuel Cycle Safety
and Safeguards
Office of Nuclear Materials Safety
and Safeguards

SUBJECT: FCSS ACTIONS TO IMPLEMENT NMSS DIRECTOR'S DECISION ON DIFFERING PROFESSIONAL VIEW CONCERNING MODELING CHEMICAL CONSEQUENCE EFFECTS FOR DETERMINING SAFETY REQUIREMENTS AT THE PROPOSED MIXED OXIDE FUEL FABRICATION FACILITY, DOCKET NUMBER 070-03098 (NMSS-DPV-2002-03)

I am responding to your memorandum dated October 3, 2003, that directed actions associated with the subject Differing Professional View (DPV).

DPV Position 1, Director's Decision 1

I request that the Division of Fuel Cycle Safety and Safeguards (FCSS) ensure that sufficient information is docketed to demonstrate the reasonableness of the Mixed Oxide (MOX) site specific application of the code results for safety related decision-making.

The FCSS staff agrees with this recommendation and based on the following information believes the appropriate actions are already completed. On June 21, 2001, staff issued a request for additional information (RAI) to Duke Cogema Stone & Webster (DCS) which requested this information. The specific request is item number 46 of this RAI (Attachment 1). DCS provided its response on August 31, 2001, including electronic copies of site-specific data that staff used to independently assess the use of ARCON96 for safety related decision-making (Attachment 2).

Based on the review of the information docketed in the DCS Construction Authorization Report (Attachment 3), the August 31, 2001, RAI response, and the FCSS independent assessment, I have confirmed that sufficient information was docketed to support the use of ARCON96. In addition, I have also confirmed that the information provided by the applicant was reviewed by

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the staff, and that the results of these evaluations provide a sufficient basis for a safety-related decision. Further, I have directed the staff preparing the final SER to ensure that the staff's review of this docketed information is thoroughly documented (Attachment 4). Copies of the reviewed information are attached.

In addition, FCSS believes it is important to address points raised in the DPV, that ". . . other applicable applicant documents have not been reviewed prior to the decision." (DPV, p. 1), or that "MOX management appeared to act in an arbitrary and capricious manner," (DPV, p. 9), or ". . . without an adequate basis, the Agency (and the applicant) gives the appearance of arbitrarily selecting a code" were not addressed. In light of the aforementioned RAI and staff evaluations of site-specific applicability: (1) the application and additional documents were reviewed prior to staff making a determination; (2) Nuclear Regulatory Commission (NRC) management ensured the results of the staff review were technically and logically sound and supported the conclusion reached, thus NRC management was not arbitrary or capricious in it's decisions; and (3) the use of the code was supported by Regulatory Guides and the staff's handbook for nuclear fuel-cycle facility accident analysis (NUREG/CR-6410) as well as thorough documentation from the applicant supporting its use.

DPV Position 2, Director's Decision 2

I request that FCSS issue guidance to ensure that its managers and staff involved with development, endorsement, use or acceptance review of automated scientific codes are familiar with relevant sections of Volume 2 of the NRC's Management Directives and NUREG/BR-0167, Software Quality Assurance Program and Guidelines.

Based on its evaluation of available review guidance, FCSS has concluded that sufficient guidance on consequence assessment is provided to the staff. This guidance is in Standard Review Plans (SRPs), such as "Standard Review Plan for the Review of License Application for a Fuel Cycle Facility" (NUREG-1520), and "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" (NUREG-1718) (Attachment 5). Both SRPs reference the "Nuclear Fuel Cycle Facility Accident Analysis Handbook" (NUREG/CR-6410) as a compendium of acceptable methods for consequence assessment, including atmospheric dispersion of airborne contaminants. In these review documents are criteria the staff can apply to endorse, use or accept scientific codes supporting fuel-cycle facility applications. Management has reviewed these documents, as part of this response, and believes it is acceptable to use them in the MOX review. It is important to note that this FCSS conclusion was supported by the Panel which "found that suitable documentation exists to guide NRC development, endorsement, and acceptance of automated scientific codes" (Sept. 30, 2003, memo to M. Virgilio, p. 2).

FCSS has also evaluated the applicability of Volume 2 of the NRC's Management Directives and NUREG/BR-0167, "Software Quality Assurance Program and Guidelines." The results of this evaluation found that neither offer guidance that was useful to the staff's review of the MOX application.

DPV Position 2, Director's Decision 3

I request that FCSS identify this for consideration in the next NMSS/RES "user-need" interface meeting.

A user-need memo to the Office of Nuclear Regulatory Research (RES) has been issued. The ADAMS Accession Number is ML033160142. The user-need memo requests assistance from RES in: (1) establishing a collaborative process involving agency stakeholders (e.g., Office of Nuclear Materials Safety and Safeguards (NMSS), Office of Nuclear Reactor Regulations (NRR) and RES) for coordinating Program Office needs for development and application of automated scientific codes used to model dispersion of the same or similar hazardous material, suitable for use for NMSS and NRR applications; (2) determining how best to inform other regulators (e.g., U.S. Environmental Protection Agency (EPA), National Oceanic and Atmospheric Administration (NOAA), Occupational Safety and Health Administration (OSHA) and U.S. Department of Energy (DOE)) and stakeholders of NRC development and application of generic scientific codes used to model dispersion of the same or similar hazardous material, when appropriate; (3) establishing an NRC public web page to inform internal and external stakeholders about NRC code work; and (4) evaluating the usefulness of the collaborative process, the effectiveness of informing other regulators, and the usefulness of the public information web page, to determine whether such activities would be useful for all generic automated scientific codes used for NRC applications. This effort should help staff determine what type of collaborative process is needed.

DPV Position 3, Director's Decision 4

I request that FCSS issue guidance so that reviewers have sufficient understanding of automated scientific codes to determine which code is appropriate (i.e., reasonable) for the intended use (e.g., providing site specific condition input for consideration in safety related decision-making). The reviewer's understanding should be sufficient enough to determine what code is appropriate (e.g., reasonable) for its intended use, its site specific application, and its results.

The response to Director's Decision 2, provides information to show that sufficient guidance is available for evaluating the acceptability of codes used to support the licensing of fuel-cycle facilities. In addition, license reviewers are technically competent by virtue of education and experience and must undergo a qualification process for their positions. The staff qualification program is a training and testing program which includes, for example, extensive instruction, on-the-job experience, self-study and, ultimately, oral review by knowledgeable and experienced staff and management. Based on the FCSS evaluation done in response to the director's decision, I have concluded that the existing qualification process ensures staff with the relevant expertise are capable of doing high-quality, technically-sound reviews in their area of expertise including selection and application of relevant cases. The criteria established in the qualification program are well defined and are applicable to appropriate technical reviewers within specific disciplines. FCSS believes the reviewer education and experience as well as the reviewer qualification program coupled with existing regulatory guidance provides a sound foundation for selection and use of codes.

Based on the actions as addressed in this memo and on the information presented, FCSS has concluded that no further actions are necessary on the Director's Decisions.

Attachments:
1. Staff RAI 46
2. DCS Response to RAI 46
3. DCS Construction Authorization Request excerpt
4. Staff's Draft SER and Draft SER, Rev 1 excerpt
5. NUREG-1718 excerpt

cc:
Robert L. O'Connell, IMNS
Margaret V. Federline, NMSS
Alexander Murray, FCSS

Attachment 1

44. Section 5.4.4.1.2, pp. 5.4-10 thru 5.4-12

Calculate the effluent concentration ratio without taking credit for the respirable fraction.

SRP Section 9.1.4.6.3.A recommends that the applicant use appropriate and verified assessment methods, computer codes, and literature values. Equation 5.4-3, the equation for 24-hour average effluent concentration ratio, contains a term for source term (ST) which is the same term as that used in Equation 5.4-2 for total effective dose equivalent to human receptors. The definition of source term, which is provided in Equation 5.4-1, has a term for the respirable fraction (RF). However, the inclusion of RF in source term derivations for demonstrating compliance with 10 CFR 70.61(c)(3) is not appropriate. This performance requirement relates to protection of the environment, not to protection of human health. Therefore, the applicant should demonstrate that the performance requirement is met for the entire range of particle sizes released to the environment, not just the respirable particle sizes.

45. Section 5.4.4.1.2, pp. 5.4-10 thru 5.4-12

Clarify how dose conversion factors from Federal Guidance Report No. 11 were chosen with due consideration for the chemical forms of radionuclides involved in accident scenarios.

Section 9.1.4.6.3.A of the SRP recommends that the applicant use appropriate and verified assessment methods, computer codes, and literature values. Section 5.4.4.1.2, "Dose Evaluation," describes the assumptions for calculating bounding total effective dose equivalent to individuals exposed during accidents, including the use of Federal Guidance Report No. 11 as the source of dose conversion factors used in the analysis. In many cases, Federal Guidance Report No. 11 provides dose conversion factors for more than one chemical form (or solubility) of the radionuclides listed. These multiple forms are represented by the transportability classes D, W and Y, where, for plutonium, the more limiting dose conversion factors are generally associated with class W compounds (such as plutonium nitrate). The application does not contain a description in Section 5.4.4.1.2 of how the solubility of various chemical forms of plutonium and americium were considered in performing the dose assessments.

46. Section 5.4.4.1.3, pp. 5.4-12 thru 5.4-13

Provide the hourly meteorological data for the period from January 1, 1987 through December 31, 1996 that was collected from the H-area meteorological tower. Include the standard deviation of the horizontal wind direction fluctuations (sigma-theta), derived stability class, wind direction, wind speed and accumulated precipitation for each hour. Include a description of how stability classes are derived using sigma-a and sigma-theta.

Section 5.4.3.2.B.v of the SRP recommends that the applicant provide a scientifically correct and reasonable estimate of the consequences from analyzed accidents. Several radiological accident consequence models that NRC may use to verify the applicant's dose calculations require hourly measurements of meteorological data. Therefore, the NRC staff must have the actual hourly data, rather than statistical summaries, to verify the correctness and reasonableness of the applicant's estimates.

Attachment 2

46. Section 5.4.4.1.3. pp. 5.4-12 thru 5.4-13

Provide the hourly meteorological data for the period from January 1, 1987 through December 31, 1996 that was collected from the H-area meteorological tower. Include the standard deviation of the horizontal wind direction fluctuations (sigma-theta), derived stability class, wind direction, wind speed and accumulated precipitation for each hour. Include a description of how stability classes are derived using sigma-a and sigma-theta.

Section 5.4.3.2.B.v of the SRP recommends that the applicant provide a scientifically correct and reasonable estimate of the consequences from analyzed accidents. Several radiological accident consequence models that NRC may use to verify the applicant's dose calculations require hourly measurements of meteorological data. Therefore, the NRC staff must have the actual hourly data, rather than statistical summaries, to verify the correctness and reasonableness of the applicant's estimates.

Response:

All downwind transport wind direction, wind speed, stability class, and sigma-theta data were extracted from two separate five-year data bases (1987 to 1991 and 1992 to 1996), which contain a complete sequential record of quality-assured SRS hourly meteorological data monitored on the H-Area meteorological tower.

All values of an empirical atmospheric turbulence intensity parameter that can be related to a Pasquill stability category are automatically derived by a microprocessor on the meteorological tower. Stability classes were based on the measured values of the standard deviation of fluctuation about the mean horizontal wind direction.

Stability categories A through G were assigned according to the range of magnitudes of sigma-theta as summarized in Table 1. Hourly precipitation data were obtained from records collected by the National Weather Service Office in Augusta, Georgia (Bush Field) and published by the National Climate Data Center (NCDC). All values of mixing height were calculated from data sets of twice daily mixing height supplied by NCDC. The daily values were determined by NCDC from a standard algorithm that used radiosonde ascents for Athens, Georgia (January 1987 through August 1994) and Atlanta, Georgia (September 1994 through December 1996), and concurrent surface data from Bush Field as input data.

The attached Meteorological Data File (on CD) contains ten files that contain hourly meteorological data used as input to the MELCOR Accident Consequence Code System (i.e., MACCS2). These files were extracted from the SRS database and contain a one-year data set for each of the years 1987 through 1996. The MACCS2 data sets consist of hourly-averaged values of plume transport sector (22.5-degree sector toward which the wind blows), wind speed (tenths of meters per second), Pasquill atmospheric stability category (1-7), and precipitation (hundredths of inches). In addition, the last line of each file gives seasonally-averaged values of morning and afternoon mixing height. Wind transport sectors in the MACCS2 files were assigned according to the transport direction ranges given in Table 2. The MACCS2 file format is described in Table 3.

The attached Meteorological Data File (on CD) contains ten files used as input into the ARCON96 computer code, one for each of the calendar years 1987 through 1996. Each file consists of a five-character station identifier (SRSOH), Julian day, hour (local time), wind direction (i.e., the direction the wind blows from) in degrees azimuth, wind speed in meters per second, and Pasquill stability classes (1-7 or A-G). All data were extracted from two separate five-year databases (1987 to 1991 and 1992 to 1996), which contain a complete sequential record of quality-assured hourly data. ARCON96 file format is described in Table 4.

The attached Meteorological Data File (on CD) contains eight files of hourly meteorological data used as input to the Environmental Protection Agency's Industrial Source Complex (ISC) atmospheric dispersion model. These files contain a one-year data set for each of the years 1989 through 1996. The ISC data sets consist of hourly values of transport direction (i.e., degrees toward which the wind blows), wind speed (meters per second), mixing height (meters), ambient temperature (degrees Kelvin), and Pasquill atmospheric stability category. The ISC file format is described in Table 5.

Action:

None

Table 1. Determination of Stability Class from Sigma-Theta Measurements

Sigma-theta* Range (degrees)	Pasquill Stability Category	Stability Identifier
Greater than 22.4	A	1
17.5-22.4	B	2
12.5-17.4	C	3
7.5-12.4	D	4
3.8-7.4	E	5
2.1-3.7	F	6
Less than 2.1	G	7

* Sigma-theta range assignments are based on criteria contained in ANSI/ANS-3.11 (2000).

Table 2. Determination of Downwind Transport Direction Sector*

Downwind Transport Direction Range (degrees)	Downwind Transport Sector	Sector Identifier
348.75-11.25	North	1
11.25-33.75	North-northeast	2
33.75-56.25	Northeast	3
56.25-78.75	East-northeast	4
78.75-101.25	East	5
101.25-123.75	East-southeast	6
123.75-146.25	Southeast	7
146.25-168.75	East-southeast	8
168.75-191.25	South	9
191.25-213.75	South-southwest	10
213.75-236.25	Southwest	11
236.25-258.75	West-southwest	12
258.75-281.25	West	13
281.25-303.75	West-northwest	14
303.75-326.25	Northwest	15
326.25-348.75	North-northwest	16

* downwind transport direction is the direction the wind is blowing toward.

Table 3. Format of the MACCS Meteorological Data Files

Column	Format	Description
2-4	I3	Julian day of the year
6-7	I2	Hour of the Day (GMT)
9-10	I2	Transport direction sector (direction wind blows toward)
11-13	I3	Wind speed (10ths of meters/sec)
14	I3	Stability Class (coded 1-7)
15-17	I3	Total precipitation (100ths of inches)

Table 4. Format of the ARCON96 Meteorological Data Files

Column	Format	Description
3-7	A5	Location identifier (SRSOH)
11-13	I3	Julian day of the year
14-15	I2	Hour of the Day (local time)
18-20	I3	Wind direction (degrees, direction from which the wind blows)
21-24	I4	Wind speed (nearest tenth of a reporting unit without the decimal, i.e., a wind speed of 5.3 m/s would be entered as 53)
26	I2	Stability class (coded 1 - 7)

Table 5. Format of the ISC Meteorological Data Files

Column	Format	Description
7-9	I3	Julian day of the year
11-12	I2	Hour of the Day (Greenwich Mean Time)
18-25	F8.0	Transport direction (degrees azimuth)
26-33	F8.2	Wind speed (meters/sec)
34-41	F8.0	Mixing height (meters)
42-49	F8.0	Temperature (Degrees Kelvin)
50-57	I8	Stability Class (coded 1-7)

Attachment 46
Additional Meteorological Data

Attached Meteorological Data File (on CD)

Attachment 3

In general, quantitative methods will be utilized for those events where risk reduction is required for the site worker or the public/environment as defined by 10 CFR §70.61. The level of risk reduction will be demonstrated to be at least equivalent to the application of qualitative methods (i.e., double contingency and/or single-failure criteria).

5.4.4 Methodology for Assessing Radiological Consequences

The methodology for assessing radiological consequences for events releasing radioactive materials is based on guidance provided in NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook* (NRC 1998b). The methodology for evaluating the consequences of a criticality event is described in Section 5.5.3.4. In this section, the methodology used to calculate radiological consequences is provided for the unmitigated and mitigated cases. Unmitigated results established from the application of this methodology are used to establish a safety strategy. Mitigated results established from the application of this methodology are presented in Section 5.5.3.

The radiological consequences for the facility worker, site worker, environment, and member of the public are assessed for events identified in the hazard evaluation. The facility worker is considered to be located near a potential accident release point. The site worker is considered to be 328 ft (100 m) from the MFFF building stack. The member of the public and the environment are considered to be located outside of the controlled area boundary approximately 5 mi (8 km) from the MFFF building stack. In the following analyses, consequences to the member of the public and environment are simply referred to as the public. Thus, limits associated with these two dose receptors, as specified in Table 5.4-1, are jointly considered when specifying event consequences.

Radiological releases are modeled as instantaneous releases to the facility worker and are conservatively modeled for the site worker and the public using a 0- to 2-hour 95th percentile dispersion χ/Q. No evacuation is credited for the assessment of the unmitigated radiological consequences.

5.4.4.1 Quantitative Unmitigated Consequence Analysis to Site Worker and Public

For each identified event sequence in the hazard evaluation, a bounding consequence for that event sequence is calculated. The bounding consequence is established by determining the applicable locations and locating the specific materials at risk from Table 5.5-2. The applicable, bounding material-at-risk values are then established from the identified values by selecting the maximum value for each form and each compound. Values for each form and compound are conservatively selected due to the dependence of the airborne release fraction, the respirable fraction, the specific activity, and the dose conversion factors.

5.4.4.1.1 Source Term Evaluation

The first step in the evaluation of the unmitigated consequences is to determine the source term. The source term is determined based on the five-factor formula as described in NUREG/CR-6410 (NRC 1998b). The five-factor formula consists of the following parameters:

- MAR – Material At Risk
- DR – Damage Ratio
- ARF – Airborne Release Fraction
- RF – Respirable Fraction
- LPF – Leak Path Factor.

These parameters are multiplied together to produce a source term (ST) representative of the amount of airborne respirable hazardous material released per a bounding scenario, as follows:

$$[ST] = [MAR] \times [DR] \times [ARF] \times [RF] \times [LPF] \quad (5.4\text{-}1)$$

Applicable, bounding quantities are established for each of these factors. Note that for entrainment events, the airborne release fraction is replaced with the airborne release rate (ARR) multiplied by the entrainment duration (i.e., ARF = ARR x duration).

The LPF in all unmitigated cases is conservatively assumed to be one (i.e., no credit is taken for leak paths). A discussion crediting LPFs in mitigated radiological consequence evaluations is provided in Section 5.4.4.3.

Applicable ARF and RF values are established for the material forms (i.e., powder, solution, pellet, rod, and filter), the material types available at the MFFF, and the release mechanisms that could potentially occur at the MFFF from values presented in NUREG/CR-6410 and DOE-HDBK-3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE 1994). Bounding ARF and RF values are then established for each material form per release mechanism by maximizing the product of these two factors of the potential material types found at the MFFF (i.e., maximizing ARF x RF for each form and per release mechanism). Thus, the result is applicable bounding ARF and RF values for specific release mechanisms for specific material forms.

For some events identified in the hazard evaluation, the identified event may encompass a number of release mechanisms. In these cases, the bounding product of the ARF and RF, per material form, will be applied to the MAR. The bounding products considered are based on the entrainment, explosive detonation, explosive overpressurization, fire/boil, and drop/crush release mechanisms for materials of a specific form.

A DR of one (1.0) is conservatively utilized to determine the radiological consequences. The sole exception is in the case of fuel rods. In this case, the DR is based upon a conservative engineering analysis of the response of structural materials for containment to the type and level of stress or force generated by the evaluated event based on available literature (e.g., SAND 1981, SAND 1987, SAND 1991).

5.4.4.1.2 Dose Evaluation

The source term is used to calculate the total effective dose equivalent (TEDE) and to establish the effluent concentration. TEDE values are calculated for exposure via the inhalation pathway to a site worker (S) and a member of the public offsite (P). Other potential pathways (e.g., submersion and ingestion) are not considered to contribute a significant fraction to the calculated

TEDE. The following expression is used to calculate the TEDE for potential radiological releases at the MFFF:

$$[\text{TEDE}]^{S,P} = [\text{ST}] \times [\chi/Q]^{S,P} \times [\text{BR}] \times [\text{C}] \times \sum_{X=1}^{N} [f]_X \times [\text{DCF}]_{effective,X} \qquad (5.4\text{-}2)$$

where:

ST	=	source term unique to each event
$[\chi/Q]^{S,P}$	=	atmospheric dispersion factor unique to the site worker and member of the public
BR	=	breathing rate
C	=	unit's conversion constant
f_x	=	includes the specific activity and the fraction of the total quantity of the MAR that is the radionuclide X
$DCF_{effective,X}$	=	effective inhalation dose conversion factor for the specified radionuclide X
N	=	total number of inhalation dose-contributing radionuclides involved in the evaluated event.

Table 5.4-3 lists the radionuclide composition of common materials located in the MFFF that have been evaluated for potential release in the hypothesized accident events.

A 24-hour average effluent concentration (EC) is calculated for a release to the environment of each of the released radionuclides using the following expression:

$$[\text{EC}]^X = \frac{[\text{ST}] \times [\chi/Q]^P \times [f]_X}{(3600 - \text{sec/hr})(24 - \text{hr})} \qquad (5.4\text{-}3)$$

Values for EC are compared to 5,000 times the values specified in Table 2 of Appendix B to 10 CFR Part 20. The ratios of the calculated value to the modified 10 CFR Part 20 value for each radionuclide are summed to ensure that the cumulative limit is satisfied, as follows:

$$\text{Total EC Ratio} = \sum_{X=1}^{N} \frac{[\text{EC}]^X}{5000 \times [\text{EC}]^X_{10CFR20}} < 1.0 \qquad (5.4\text{-}4)$$

Atmospheric dispersion factors (χ/Q) for the site worker and a member of the public were established from SRS data using the MACCS2 and ARCON96 computer codes. These codes are briefly discussed in Section 5.4.4.1.3.

The breathing rate (BR) is conservatively assumed to be 3.47×10^{-4} m^3/sec (20.8 L/min). This value is from Regulatory Guide 1.25 (NRC 1972) and is equivalent to the uptake volume (10 m^3)

of a worker in an 8-hour workday. The inhalation dose conversion factors (DCFs) are taken from Federal Guidance Report No. 11 (EPA 1989).

Once unmitigated radiological consequences (TEDE and EC) are established for each event identified in the hazard assessment, events are grouped and bounding events are established for each of these groupings under each event type. Unmitigated radiological consequences established for each bounding event are then compared to the limits in Table 5.4-1. Based on this comparison and potential prevention and/or mitigation features available to each event grouping, the safety strategy is established for each bounding event within an event type.

5.4.4.1.3 Atmospheric Dispersion Evaluation

5.4.4.1.3.1 MACCS2

The MACCS2 (MELCOR Accident Consequence Code System for the Calculation of the Health and Economic Consequences of Accidental Atmospheric Radiological Releases) computer code was used to compute the downwind relative air concentrations (χ/Q) for a 1-hour ground-level release from the MFFF. The relative concentration (atmospheric dispersion factors) (χ/Q) is the dilution provided relative to site meteorology and distance to the receptor(s). MACCS2 simulates the impact of accidental atmospheric releases of radioactive materials on the surrounding environment. A detailed description of the MACCS2 model is available in NUREG/CR-6613 (NRC 1998a).

A MACCS2 calculation consists of three phases: input processing and validation, phenomenological modeling, and output processing. The phenomenological models are based mostly on empirical data, and the solutions they entail are usually analytical in nature and computationally straightforward. The modeling phase is subdivided into three modules. ATMOS treats atmospheric transport and dispersion of material and its deposition from the air utilizing a Gaussian plume model with Pasquill-Gifford dispersion parameters. EARLY models consequences of the accident to the surrounding area during an emergency action period. CHRONIC considers the long-term impact in the period subsequent to the emergency action period.

The receptor of interest includes the maximally exposed offsite individual (MOI) at 5 mi (8 km). The input into the MACCS2 code included SRS meteorological data files. The SRS meteorological data files are composed of hourly data for SRS for each calendar year from 1987 through 1996. No credit is taken for building wake effects. The release is assumed to be from ground level at the MFFF, without sensible heat, over 1 hour. For conservatism, no wet or dry deposition has been assumed.

The dose incurred by the MOI is reported at the 95th percentile level without regard to sector. The MOI is assumed to be located at the closest site boundary, which is 5 mi (8 km) from the MFFF.

5.4.4.1.3.2 ARCON96

The ARCON96 computer code was used to compute the downwind relative air concentrations (χ/Q) for the onsite receptor located within 328 ft (100 m) of a ground-level release from the MFFF to account for low wind meander and building wake effects. ARCON96 implements a straight-line Gaussian dispersion model with dispersion coefficients that are modified to account for low wind meander and building wake effects (NRC 1997). A constant release rate is assumed for the entire period of release. Building wake effects are considered in the evaluation of relative concentration from ground-level releases. ARCON96 calculates relative concentration using hourly meteorological data. It then combines the hourly averages to estimate concentrations for periods ranging in duration from 2 hours to 30 days. Wind direction is considered as the averages are formed. As a result, the averages account for persistence in both diffusion conditions and wind direction. Cumulative frequency distributions are prepared from the average relative concentrations. Relative concentrations that are exceeded no more than 5% of the time (95^{th} percentile relative concentrations) are determined from the cumulative frequency distributions for each averaging period.

Atmospheric dispersion factors (χ/Q) for ground-level releases to the site worker and a member of the public were established using these codes as 4.2×10^{-4} sec/m^3 and 3.7×10^{-6} sec/m^3, respectively.

5.4.4.2 Consequence Analysis for the Facility Worker

For the facility worker, conservative consequences are qualitatively estimated. The facility worker is assumed to be at the location of the release. Thus, for events evaluated in the preliminary accident analysis involving an airborne release of plutonium or americium, principal SSCs are deterministically applied. For events involving the release of uranium, the unmitigated consequences are estimated to be low and principal SSCs are not applied.

5.4.4.3 Quantitative Mitigated Consequence Analysis

The methodology used to establish the mitigated radiological consequences closely follows the methodology used to establish the unmitigated consequences. Mitigated consequences are calculated for those bounding events representing an event grouping in which mitigation features will be utilized to reduce the risk in accordance with 10 CFR §70.61.

To perform the mitigated consequence analysis, the consequence analysis methodology described in the previous section is utilized with the following modification: applicable bounding LPF values are used for the principal SSCs providing mitigation. This LPF is associated with the fraction of the radionuclides in the aerosol that are transported through some confinement deposition or filtration mechanism. There can be many LPFs for some events, and their cumulative effect is often expressed as one value that is the product of all leak path multiples. Inclusion of these multiples in a single LPF is done to clearly differentiate between calculations of doses without mitigation (where the LPF is assumed equal to one) and calculations of doses with mitigation (where the LPF reflects the dose credit provided to the controls). In this manner, the LPF represents the credit taken for the mitigating principal SSCs at the MFFF.

- Analysis of failure modes and common mode failures
- Special inspection, testing, and maintenance requirements
- Management measures applied to the item and the basis for grading
- Safety parameters controlled by the item, safety limit on the parameter
- Assessment of the impact of non-safety features on IROFS ability to perform their function.

These analyses will be applied to each event sequence with the potential to exceed 10 CFR §70.61 requirements. The analyses verify that single failure criterion or double contingency principle is effectively applied, that there are no common mode failures, that the IROFS will be effective in performing their intended safety function, that the conditions that the IROFS will be subjected to will not diminish the reliability of the IROFS, and also identify and verify appropriate IROFS failure detection methods. Each of the event sequences and the accompanying specific measures provided by the aforementioned deterministic criteria will be documented in the ISA and summarized in the ISA summary. This combination of analyses will demonstrate that the likelihood requirements of 10CFR70.61 are satisfied.

In conjunction with (but separate from) the safety/licensing basis to provide additional confidence in the demonstration of the adequacy of these deterministic design criteria, a supplemental likelihood assessment will be conducted for events (excluding NPH events) that could result in consequences that exceed the threshold criteria for the site worker or the public. This supplemental assessment will be based on the guidance provided in NUREG 1718 and will demonstrate a target likelihood comparable to a "score" or -5 as defined in Appendix A of NUREG 1718.

5.4.4 Methodology for Assessing Radiological Consequences

The methodology for assessing radiological consequences for events releasing radioactive materials is based on guidance provided in NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook* (NRC 1998b). The methodology for evaluating the consequences of a criticality event is described in Section 5.5.3.4. In this section, the methodology used to calculate radiological consequences is provided for the unmitigated and mitigated cases. Unmitigated results established from the application of this methodology are used to establish a safety strategy. Mitigated results established from the application of this methodology are presented in Section 5.5.3.

The radiological consequences for the facility worker, site worker, member of the public, and the environment are assessed for events identified in the hazard evaluation. The facility worker is considered to be within the MFFF located inside a room near a potential accident release point. The site worker is considered to be 328 ft (100 m) from the MFFF building stack. The member of the public is considered to be located near the controlled area boundary at approximately 5 mi

(8 km) from the MFFF building stack. The controlled area is defined as an area outside of a restricted area but inside the site boundary to which access can be limited by the licensee for any reason. The nearest site boundary is 5.4 miles (8.8 km) and the nearest SRS controlled access point is 5.1 miles (8.1 km). A restricted area is an area to which access is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. The MFFF restricted area is coincident with the protected area, an area encompassed by physical barriers and to which access is controlled and is located at 170.6 ft (52 m) from the MFFF building stack. Radiological consequences to the environment -are assessed outside the MFFF restricted area (i.e., at the Restricted Area Boundary).

Radiological releases are modeled as instantaneous releases to the facility worker and are conservatively modeled for the site worker, the public, and the environment using a 0- to 2-hour 95th percentile dispersion χ/Q. No evacuation is credited for the assessment of the unmitigated radiological consequences.

5.4.4.1 Quantitative Unmitigated Consequence Analysis to Site Worker and Public

For each identified event sequence in the hazard evaluation, a bounding consequence for that event sequence is calculated. The bounding consequence is established by determining the applicable locations and locating the specific materials at risk from Tables 5.5-3a and 5.5-3b. The applicable, bounding material-at-risk values are then established from the identified values by selecting the maximum value for each form and each compound. Values for each form and compound are conservatively selected due to the dependence of the airborne release fraction, the respirable fraction, the specific activity, and the dose conversion factors.

5.4.4.1.1 Source Term Evaluation

The first step in the evaluation of the unmitigated consequences is to determine the source term. The source term is determined based on the five-factor formula as described in NUREG/CR-6410 (NRC 1998b). The five-factor formula consists of the following parameters:

- MAR – Material At Risk
- DR – Damage Ratio
- ARF – Airborne Release Fraction
- RF – Respirable Fraction
- LPF – Leak Path Factor.

These parameters are multiplied together to produce a source term (ST) representative of the amount of airborne respirable hazardous material released per a bounding scenario, as follows:

$$[ST] = [MAR] \times [DR] \times [ARF] \times [RF] \times [LPF] \qquad (5.4\text{-}1)$$

Applicable, bounding quantities are established for each of these factors. Note that for entrainment events, the airborne release fraction is replaced with the airborne release rate (ARR) multiplied by the entrainment duration (i.e., ARF = ARR x duration). It has been assumed that the duration of the entrainment release is one hour, assuming no evacuation. The unmitigated consequences associated with entrainment events are orders of magnitude below those associated with the bounding events. A longer duration of release up to the entire MAR involved in the event would not impact the safety strategy and the mitigated consequences would still be acceptable.

The LPF in all unmitigated cases is conservatively assumed to be one (i.e., no credit is taken for leak paths). A discussion crediting LPFs in mitigated radiological consequence evaluations is provided in Section 5.4.4.4.

Applicable ARF and RF values are established for the material forms (i.e., powder, solution, pellet, rod, and filter), the material types available at the MFFF, and the release mechanisms that could potentially occur at the MFFF from values presented in NUREG/CR-6410 and DOE-HDBK-3010, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE 1994). Bounding ARF and RF values are then established for each material form per release mechanism by maximizing the product of these two factors of the potential material types found at the MFFF (i.e., maximizing ARF x RF for each form and per release mechanism). Thus, the result is applicable bounding ARF and RF values for specific release mechanisms for specific material forms.

For some events identified in the hazard evaluation, the identified event may encompass a number of release mechanisms. In these cases, the bounding product of the ARF and RF, per material form, will be applied to the MAR. The bounding products considered are based on the entrainment, explosive detonation, explosive overpressurization, fire/boil, and drop/crush release mechanisms for materials of a specific form.

A DR of one (1.0) is conservatively utilized to determine the radiological consequences for most material forms and events. Exceptions include fuel rods and pellets for an explosive over-pressurization event, fires in select storage areas, and the drop of fuel assemblies.

5.4.4.1.2 Dose Evaluation

The source term is used to calculate the total effective dose equivalent (TEDE). TEDE values are calculated for exposure via the inhalation pathway to a site worker (S) and a member of the public offsite (P). Other potential pathways (e.g., submersion and ingestion) are not considered to contribute a significant fraction to the calculated TEDE. The following expression is used to calculate the TEDE for potential radiological releases at the MFFF:

$$[TEDE]^{S,P} = [ST] \times [\chi/Q]^{S,P} \times [BR] \times [C] \times \sum_{X=1}^{N} [f]_X \times [DCF]_{effective,X} \qquad (5.4\text{-}2)$$

where:

ST	=	source term unique to each event
$[\chi/Q]^{S,P}$	=	atmospheric dispersion factor unique to the site worker and member of the public
BR	=	breathing rate
C	=	unit's conversion constant

f_x	=	includes the specific activity and the fraction of the total quantity of the MAR that is the radionuclide X
$DCF_{effective,X}$	=	effective inhalation dose conversion factor for the specified radionuclide X
N	=	total number of inhalation dose-contributing radionuclides involved in the evaluated event.

Table 5.4-3 lists the radionuclide composition of common materials located in the MFFF that have been evaluated for potential release in the hypothesized accident events.

Atmospheric dispersion factors (χ/Q) for the site worker and a member of the public were established from SRS data using the MACCS2 and ARCON96 computer codes. These codes are briefly discussed in Section 5.4.4.1.3.

The breathing rate (BR) is conservatively assumed to be 3.47×10^{-4} m^3/sec (20.8 L/min). This value is from Regulatory Guide 1.25 (NRC 1972) and is equivalent to the uptake volume (10 m^3) of a worker in an 8-hour workday.

The inhalation dose conversion factors (DCFs) are taken from Federal Guidance Report No. 11 (EPA 1989), based on the form of the potential releases from the MFFF when received by the dose receptor. For the MFFF, dose receptors are conservatively assumed exposed to oxides of unpolished plutonium, polished plutonium, and/or uranium, and/or elemental americium. The oxides have specific activities (molecular) that are greater by a factor of 2 than those of other potential release forms (e.g., plutonium oxalates and nitrates). For many radionuclides, Federal Guidance Report No. 11 provides dose conversion factors for more than one chemical form (or solubility). The multiple forms are represented by transportability classes. For the MFFF, Y class DCFs have been used for all radionuclides except americium, which only has a W class DCF. Releases of soluble materials are bounded by those of the insoluble form because the amount of MAR in the bounding events for soluble releases is smaller than the amount of MAR for the insoluble releases.

Once unmitigated radiological consequences are established for each event identified in the hazard assessment, events are grouped and bounding events are established for each of these groupings under each event type. Unmitigated radiological consequences established for each bounding event are then compared to the limits in Table 5.4-1. Based on this comparison and potential prevention and/or mitigation features available to each event grouping, the safety strategy is established for each bounding event within an event type.

5.4.4.1.3 Atmospheric Dispersion Evaluation

5.4.4.1.3.1 MACCS2

The MACCS2 (MELCOR Accident Consequence Code System for the Calculation of the Health and Economic Consequences of Accidental Atmospheric Radiological Releases) computer code was used to compute the downwind relative air concentrations (χ/Q) for a 1-hour ground-level release from the MFFF. The relative concentration (atmospheric dispersion factors) (χ/Q) is the dilution provided relative to site meteorology, elevation of release, and distance to the receptor(s). MACCS2 simulates the impact of accidental atmospheric releases of radioactive materials on the surrounding environment. A detailed description of the MACCS2 model is available in NUREG/CR-6613 (NRC 1998a).

A MACCS2 calculation consists of three phases: input processing and validation, phenomenological modeling, and output processing. The phenomenological models are based mostly on empirical data, and the solutions they entail are usually analytical in nature and computationally straightforward. The modeling phase is subdivided into three modules. ATMOS treats atmospheric transport and dispersion of material and its deposition from the air utilizing a Gaussian plume model with Pasquill-Gifford dispersion parameters. EARLY models consequences of the accident to the surrounding area during an emergency action period. CHRONIC considers the long-term impact in the period subsequent to the emergency action period.

The receptor of interest includes the maximally exposed offsite individual (MOI) at the controlled area boundary. The input into the MACCS2 code included SRS meteorological data files. The SRS meteorological data files are composed of hourly data for SRS for each calendar year from 1987 through 1996. No credit is taken for building wake effects. The release is assumed to be from ground level at the MFFF, without sensible heat, over 1 hour. For conservatism, no wet or dry deposition has been assumed.

The dose incurred by the MOI is reported at the 95th percentile level without regard to sector. The MOI is assumed to be located at the closest site boundary to the MFFF. The one-hour atmospheric dispersion factor (χ/Q) for ground-level releases to a member of the public located at the controlled area boundary (approximately 5 mi [8 km] from the MFFF stack) was computed by MACCS2 to be 3.7×10^{-6} sec/m^3.

5.4.4.1.3.2 ARCON96

The ARCON96 computer code was used to compute the downwind relative air concentrations (χ/Q) for the site worker located within 328 ft (100 m) of a ground-level release from the MFFF to account for low wind meander and building wake effects.

ARCON96 implements a normal straight-line Gaussian dispersion model with dispersion coefficients that are empirically modified from atmospheric tracer and wind tunnel experimental data to account for low wind meander and aerodynamic effects of buildings on the near-field wind field (e.g., wake and cavity regions) (NRC 1997). Hourly, normalized concentrations (χ/Qs) are calculated from hourly-averaged meteorological data. The hourly values are averaged to develop χ/Qs for five periods ranging from 2 to 720 (i.e., 0 to 2 hr, 2 to 8 hr, 8 to 24 hr, 1 to 4 days, and 4 to 30 days) hours in duration. Of these time periods, only the 0 to 2 hr interval is used for dose calculations. ARCON96 accounts for wind direction as the averages are formed. To ensure that the most conservative χ/Q was selected for dose calculations, χ/Q determinations were made for 16 different wind directions. As a result, the averages account for persistence in both diffusion conditions and wind direction. Cumulative frequency distributions are prepared from the average relative concentrations. Relative concentrations that are exceeded no more than

5% of the time (i.e., 95th percentile relative concentrations) are determined from the cumulative frequency distributions for each averaging period.

The two-hour atmospheric dispersion factor (χ/Q) for ground-level releases to the site worker at 328 ft (100 m) was calculated by ARCON96 to be 6.1×10^{-4} sec/m^3.

5.4.4.2 Consequence Analysis for the Facility Worker

For the facility worker, conservative consequences are qualitatively estimated. The facility worker is assumed to be at the location of the release. Thus, for events evaluated in the preliminary accident analysis involving an airborne release of plutonium or americium, principal SSCs are deterministically applied. For events involving the release of uranium, the unmitigated consequences are estimated to be low and principal SSCs are not applied.

5.4.4.3 Environmental Consequences

A 24-hour average effluent concentration (EC) is calculated for a release to the environment of each of the released radionuclides using the following expression:

$$[EC]^X = \frac{[ST]/[RF] \times [\chi/Q]^{RA} \times [f]_X}{(3600 - sec/hr)(24 - hr)} \qquad (5.4\text{-}3)$$

where:

$[\chi/Q]^{RA}$ = atmospheric dispersion factor unique to the restricted area boundary

The 24-hour average atmospheric dispersion factor $(\chi/Q)^{RA}$ for ground-level releases at the restricted area boundary (171 ft [52 m]) was calculated to be 2.79×10^{-04} sec/m^3 by ARCON96.

Since the radiological consequences to the environment are limited to an airborne effluent concentration and not a respirable quantity, the respirable fraction (RF) in Equation 5.4-3 corrects the source term (Equation 5.4-2) such that the source term reflects an airborne quantity.

Table 5.4-3 lists the radionuclide composition of common materials located in the MFFF that have been evaluated for potential release in the hypothesized accident events.

Values for EC are compared to 5,000 times the values specified in Table 2 of Appendix B to 10 CFR Part 20, which are listed in Table 5.4-3. The ratios of the calculated value to the modified 10 CFR Part 20 value for each radionuclide are summed to ensure that the cumulative limit is satisfied, as follows:

$$\text{Total EC Ratio} = \sum_{X=1}^{N} \frac{[EC]^X}{5000 \times [EC]^X_{10CFR20}} < 1.0 \qquad (5.4\text{-}4)$$

Once unmitigated environmental consequences are established for each event identified in the hazard assessment, events are grouped, and bounding events are established for each of these groupings under each event type. Unmitigated environmental consequences established for each

Attachment 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," DSER Section 9.1.1.4.4).

As described in Section 11.4 of this 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. Appendix F of Reference 9.3.12 recommends an efficiency of between 99 percent and 95 percent. Therefore, for the purposes of this DSER, the staff has reanalyzed the accident consequences for fire and explosion events using an leak path factor (LPF) of 0.01. The applicant disagrees with the staff on this point. Therefore, the methodology for deriving source terms that was presented by the applicant has not been accepted by the staff and is considered an open issue.

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. The staff independently verified the applicant's use of these guides for estimating source terms, and find the applicant's analysis to be consistent with the guidance.

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 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 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):

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

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 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 4.2E-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.

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

Attachment 5

c. Explains for processes vulnerable to criticality accidents, why it is expected that the given design and design bases will meet the double contingency requirement of 10 CFR 70.64(a)(9).

As discussed in Item iii below, the accident consequences will depend on the design bases of the principal SSCs. When analyzing accident sequences, the applicant should examine the failure of ALL features, structures, control devices, equipment, or procedures to ensure that all principal SSCs are appropriately identified.

iii. Consequence assessment

The applicant's consequence assessment is sufficiently quantitative to compare the consequence estimates against the performance requirements of 10 CFR 70.61. The applicant does not determine the consequences for all accidents and all SSCs individually; however, the applicant demonstrates that the consequence assessment is bounding through the applicant's analysis of representative processes sufficient to cover all principal types of hazardous materials.

iv. Likelihood Assessment

The applicant provides information that indicates that the frequencies of accidents are in accordance with the acceptance criteria for the applicant's likelihood definitions. The applicant's safety assessment of the design bases with respect to likelihood provides reasonable assurance that the likelihood requirements of 10 CFR 70.61 will be met by the final design. The applicant commits to using equivalent or refined definitions of likely, unlikely, highly unlikely, and credible in the ISA. In addition, the applicant describes the likelihood evaluation method to be used in the ISA. The applicant makes these methods and definitions part of the design bases.

F. The applicant describes the principal SSCs. This description should include:

i. The number, types, and description of the principal SSCs. In particular, the applicant describes the general features that indicate that the principal SSCs can be designed and constructed to meet the design bases.

The description of the principal SSCs need not be at the level of detailed engineering drawings. However, principal safety function features, devices, amounts of hazardous materials, and the principal dimensions, layout, and location relevant to safety must be given. Each general type of principal SSC or process using the same design bases must be described. However, approximate numbers of each general type of principal SSC or process is sufficient. It is the safety basis that is to be assessed.

ii. For each principal SSC, the parameters that will be specified or controlled for safety and the ranges and values of those parameters that constitute the design bases. For active engineered controls, the applicant states the type of sensing and the type of control device. For passive engineered controls, the applicant states the general geometry,