USEC
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May 23, 2005
AET 05-0038

Mr. Jack R. Strosnider
Director, Office of Nuclear Material Safety and Safeguards
Attention: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

American Centrifuge Plant
Docket Number 70-7004
Submittal of Additional Information Related to Request for Additional Information – License Application for the American Centrifuge Plant (TAC Nos. L32306, L32307, and L32308)

Dear Mr. Strosnider:

USEC Inc. (USEC) hereby submits to the U.S. Nuclear Regulatory Commission (NRC) additional information related to Attachments 1 and 2 of Reference 1.

Enclosure 1 to this letter provides additional information related to Criticality Safety. Enclosure 2 to this letter provides additional information related to Decommissioning Funding. Enclosure 3 to this letter provides the non-proprietary changed pages for the License Application and supporting documents identified in Reference 2. Changes from the previous revision submitted to the NRC are designated with revision bars in the right hand margin. As discussed in response to item #1 (NRC RAI CA-3) of the Criticality Safety topic, Enclosure 4 to this letter provides the reference document K-D-1987 entitled "Water Immersion Tests of UF_6 Cylinders With Simulated Damage."

Enclosures 1 through 4 have been determined, in accordance with the guidance provided by the U.S. Department of Energy, to not contain Export Controlled Information. Enclosures 1 through 3 have been reviewed in accordance with the December 21, 2004 NRC Review Criteria to Identify Sensitive Information in Fuel Cycle Documents and the appropriate changes are being withheld and have been submitted to the NRC under separate cover (AET 05-0037).

NMSS01

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If you have any questions regarding this matter, please contact Peter J. Miner at (301) 564-3470.

Sincerely,

S. A. Toelle

Steven A. Toelle
Director, Nuclear Regulatory Affairs

cc: Y. Faraz, NRC HQ
B. Smith, NRC HQ

Enclosures: As Stated

Reference:

1. NRC memorandum to James W. Clifford (NRC) from Yawar Faraz (NRC) regarding April 25, 2005 Meeting Summary: USEC Inc. Response to NRC Request for Additional Information," dated May 9, 2005.

2. USEC Letter AET 05-0037 from S.A. Toelle (USEC) to J.R. Strosnider (NRC), "Submittal of Additional Information Related to Request for Additional Information – License Application for the American Centrifuge Plant (TAC Nos. L32306, L32307, and L32308) – Proprietary and Export Controlled Information," dated May 19, 2005.

Enclosure 1 to AET 05-0038

Responses to Attachment 2 of the April 25, 2005 Meeting Summary (Criticality Safety)

Enclosure 1 of AET 05-0038

1. NRC's RAI NC-7 requested the applicant to clarify whether postings and/or labels were required for administrative controls whenever an in-hand procedure was not used. The applicant responded that postings and/or labels are not required when using in-hand procedures, but that they may still be used on occasion.

   The NRC staff indicated that this did not respond to the question about what was done when in-hand procedures are not used.

   The applicant stated that it would document any instances in which postings were not considered necessary on a case-by-case basis, and that it would have similar written criteria for the placement of postings to what it has currently for the gaseous diffusion plants (GDPs). Postings would be most appropriate for areas in which operations were localized to small areas with limited operations. The NRC staff indicated that it would consider the specific implementation of posting requirements appropriate to review during the operational readiness review (ORR) that the NRC will conduct after any license is issued and prior to the commencement of operations.

<u>USEC Response</u>

Postings and/or labels are not required when an in-hand procedure is used. However, postings and/or labels may still be used to ensure adequate dissemination of Nuclear Criticality Safety (NCS) requirements to all personnel. The application of posting and/or labels for all administrative NCS controls is determined through a documented process with the intent of disseminating necessary information to personnel in the field, while also not overloading personnel with excessive visual clutter that dilutes the effectiveness of the postings. Therefore, field postings and/or labels may not be required for operations that do not occur in a well-defined area or for operations that are controlled from a central control point. Administrative controls that are incorporated into a formalized tracking system for surveillances, calibrations, or periodic maintenance may not need to be posted. Administrative controls that are purely administrative (e.g., completing a log book, hanging lock out/tag out tags, etc.) may not need to be posted. The fissile material operations manager and the cognizant NCS engineer may also determine together that a specific control does not require posting for some other reason and this determination and the reason is documented. The required postings are described in the Nuclear Criticality Safety Evaluation (NCSE) and are approved by the Plant Safety Review Committee as described in Chapter 5.0 of the License Application.

When neither postings nor in-hand procedures are used, the knowledge and training of personnel is relied upon to maintain compliance with NCS controls. Fissile material operations at the ACP will be performed through either procedures or work instructions. Whether the procedures are in-hand or not, procedure compliance is required. Personnel who are not trained on fissile material operations or are not directed to perform a fissile material operation will not independently or spontaneously perform fissile work. Moreover, fissile material operations are performed according to written procedures or

work instructions, not according to a posting. The presence of a posting, therefore, is to provide direction to the untrained, casual, passerby, to prevent them from inadvertently adversely affecting NCS controls.

Based on experience with the gaseous diffusion plant (GDP), operations like this are operations where inadvertently adversely affecting NCS controls would require conscious action. For example, removed GDP equipment is required to maintain spacing from other removed equipment. Given the hundreds of components spread across dozens of acres of floor space, it would be impractical to post the spacing requirements for each item. However, the large size of the equipment (hundreds or thousands of pounds) prevents inadvertent movement by untrained personnel. Therefore, for this example, posting spacing requirements is unnecessary.

No changes to the License Application are required.

2. NRC's RAI NC-10 requested the applicant to justify why annual walkthroughs of its fissile material operations are acceptable. The applicant responded that front-line managers (FLMs) provide real-time assessment of fissile material operations (FMOs) to ensure that NCS requirements are being adequately implemented.

The NRC staff indicated that it was not clear whether FLMs are procedurally required to monitor NCS compliance among their other responsibilities or whether they are adequately trained to do so.

The applicant stated that there would be NCS staff on the floor regularly, and that the FLMs would be provided with additional training in NCS sufficient to allow them to perform this duty. In addition, while there is a commitment to walk down every area at least annually, NCS staff would be expected to walk down portions of the facility on approximately a monthly basis. The NRC staff stated that it was initially unclear whether the annual walkdown meant that every area of the facility would be walked down at least once per year. The applicant agreed to clarify the intent and provide additional details on training of FLMs.

<u>USEC Response</u>

Section 5.3.4 of the License Application has been revised to clarify the intent of the NCS walk-through program and include a reference to Section 11.3.1.4 for training of crew supervision. The revision of Section 5.3.4 states: "As a minimum, specific fissile material operating areas are assessed by NCS personnel via walk-through at least annually, sometimes in conjunction with the assessments discussed below. By distributing the various areas' walk-throughs over a year's time, NCS personnel are performing a field walk-through on approximately a monthly basis."

Additionally, Section 5.3.4 was revised to state, "Managers in charge of fissile material operations are provided additional training on NCS and response to NCS deficiencies as described in Section 11.3.1.4 of this license application."

3. NRC's RAI NC-11 requested the applicant to state how often NCS Program audits would be performed. The applicant responded that NCS Program audits would be performed at least once every three years.

The NRC staff indicated that the applicant had not adequately justified why a triennial frequency was acceptable.

The applicant stated that, while external and independent audits of the NCS program would be performed on a triennial basis, there would also be an annual self-assessment of the NCS program, which is consistent with the self-assessment frequency for other safety programs. The applicant agreed to clarify this.

<u>USEC Response</u>

Section 5.3.4 of the License Application has been revised to include a commitment to perform an annual organizational self-assessment of their respective implementation of the NCS program. The revision to Section 5.3.4 states: "Fissile material operations management also performs an annual self-assessment to ensure NCS program requirements are being met in the field. In addition to the annual self-assessments, independent internal audits of the NCS Program are conducted or coordinated by the Quality Assurance Manager as described in Section 11.5 of this license application."

4. NRC's RAI NC-12 requested the applicant to clarify when a nuclear criticality safety evaluation (NCSE) was necessary. The applicant responded that there were a variety of situations in which requests for NCS evaluations were not necessary, including: (1) the operation is obviously non-fissile, (2) the operation has not been funded, (3) the operation lies within the scope of an existing NCSE, (4) the operation is non-fissile due to limited inventory or enrichment, or (5) the operation cannot be shown to be doubly contingent or highly unlikely and therefore would not be approved.

The NRC staff indicated that these situations appeared reasonable but that the response was open-ended in that it did not provide comprehensive criteria addressing when an NCSE was not required and therefore cannot be approved.

The applicant stated that historically, in the experience of the GDPs, some operations had been overly conservative in requesting NCS evaluations and that there were many different instances in which they were not needed. The applicant also clarified that any operations exceeding 100 g ^{235}U or an enrichment of 1 wt% ^{235}U under either normal or credible abnormal conditions would be covered by an NCSE. The NRC staff stated that this would be adequate and no further response was needed beyond a written confirmation of USEC's statements made during the meeting as reflected above.

USEC Response

As described in Section 5.4.2 of the License Application, NCSEs are only required for operations that involve 100 grams (g) ^{235}U or more at 1 weight (wt.) percent or more ^{235}U enrichment. USEC confirms the statements made during the meeting as reflected above.

5. NRC's RAI NC-13 requested the applicant to describe how reliance is placed on the natural and credible course of events, and what "other means" may be used besides the use of administrative controls, engineered controls, and the natural and credible course of events. The applicant responded that the natural and credible course of events was relied on when implementing explicit NCS controls is unnecessary, impractical, or overly prescriptive. To qualify as the natural and credible course of events, no controls must be necessary to maintain them.

The NRC staff indicated that the response did not justify why this type of event could be credited without establishing specific controls to maintain initial conditions, and did not address the question about use of "other means" of criticality control.

The applicant stated that its position on use of the natural and credible course of events was consistent with the approach previously found acceptable between the NRC and USEC for the GDPs. The applicant agreed to provide the NRC staff with documentation of this agreement for its review. The applicant also stated that it included use of the words "other means" from American National Standards Institute/American Nuclear Society (ANSI/ANS) 8.1 for completeness because it could not envision all possible types of situations that may arise in the future, and that whatever means the applicant uses, it would still need to ensure that criticality is highly unlikely and that the operation is doubly contingent. The applicant further stated that it does not currently use "other means," and that any change that resulted in their use would necessarily require NRC pre-approval, because it would constitute a new accident sequence or change to an existing accident sequence under 10 CFR 70.72. The NRC staff stated that this would be acceptable as it could then make a determination of adequacy of these controls on a case-by-case basis.

USEC Response

As discussed in the summary above, this issue has been addressed with the NRC Staff and changes to the GDP Safety Analysis Report were made to describe reliance on the "natural and credible course of events." Information was provided to the NRC in correspondence dated January 22, 2001, (letter GDP 01-2003). The ACP plans to follow the same practice outlined in this letter and included a discussion in Section 5.4.2 of the License Application. No changes to the License Application are required.

6. NRC's RAI NC-14 requested the applicant to describe how it will ascertain whether a change in process conditions is sufficiently "unlikely" to meet the double contingency principle (DCP). The applicant responded that the criterion for determining when process conditions are sufficiently unlikely to meet the DCP is engineering judgment, combined with a peer review and the rest of the NCSE approval process.

The NRC staff indicated that citing engineering judgment alone is too subjective and that more specificity was needed.

The applicant stated that the FLMs were required to check criticality controls to ensure that they were capable of being implemented and performing their intended safety functions. In addition, the applicant cited the many layers of reviews and available operating history to provide assurance that double contingency controls would be unlikely to fail. The NRC staff indicated that the reliance on engineering judgment and the approval process was subjective and did not provide any objective criteria that could be used to unambiguously determine whether the failure of the controls was in fact unlikely. The applicant stated that it would revise its response but would discuss it with the NRC before responding.

USEC Response

The criteria used to determine whether a proposed control is sufficiently unlikely to meet the DCP is engineering judgment, combined with a peer review and the rigors of the NCSE approval process. When available, actual operating history or failure frequencies are used. The conclusion of "unlikely" must be shared by the NCS Engineer who authors the NCSE, the Senior NCS Engineer who peer reviews the NCSE, and the NCS Manager who approves the NCSE. The development of the criticality accident scenarios, selection of controls, and determination of postings all have input from personnel who will be performing or managing the operation. The personnel providing input may be the specific operators or maintenance personnel doing the work, their direct line supervision, or other management directly responsible for that work. Approval of the NCSE by the fissile material operations manager is done by a manager who has the authority and responsibility to ensure that procedures will be modified, personnel trained, and other implementation steps are completed prior to initiation of the operation. By approving the NCSE the fissile material operations manager acknowledges agreement with the controls and acceptance of the limits imposed.

The NCSE will document the basis for the conclusion that a change in a process or parameter is "unlikely." The basis may be an engineered feature, administrative control, the natural or credible course of events, or any combination of these or other means necessary to ensure the change is unlikely to occur. The parameters or conditions relied on and the limits must be specified in the NCSE and controlled.

Separate from the process to develop and implement NCSEs to comply with the double contingency principle, the Integrated Safety Analysis reviewed credible criticality accident scenarios. For those scenarios that were determined to be credible paths to a

criticality accident, sufficient IROFS were developed to ensure the likelihood of a criticality accident was highly unlikely in accordance with the performance criteria of 10 *Code of Federal Regulation* (CFR) 70.61.

The combination of compliance with the double contingency principle and the performance requirements of 10 CFR 70.61 helps to ensure that criticality accident scenarios are very tightly controlled. Reliance on engineering judgment is the foundation of all of these hazard analysis techniques that rely on the experience and knowledge of subject matter experts in their field. For the ACP, each of the hazard analysis efforts, including NCS, involved personnel familiar with uranium enrichment, UF_6 handling techniques, GDP operations, and GDP history. The personnel performing the NCS analyses were all qualified Senior NCS Engineers familiar with analyzing both HEU and LEU uranium systems. Most of the NCS personnel had extensive experience at either the Portsmouth and/or Paducah GDP.

No single engineer's judgment forms the basis for any conclusion of "unlikely" for any specific control. Management systems ensure that each double contingency control is reviewed and accepted by a diverse group of individuals whose expertise in their fields gives them the clarity and independence to critically examine each double contingency control to ensure it is an available and reliable control. No changes to the License Application are required.

7. NRC's RAI NC-15 requested the applicant to clarify the meaning of the paragraph on page 5-9 discussing "items related to NCS." The applicant responded that "items related to NCS" meant either (a) items maintained by non-NCS programs (e.g., Health Physics, Industrial Safety); or (b) chains of events that are inherently unlikely without specifying explicit controls.

The NRC staff indicated that the examples did not support the statement that items related to NCS do not require explicit controls. In the first example, that of five different valves that must be misaligned, the fact that there are five valves is a feature of the design that must be recognized as important for NCS. In the second example, the threat of imminent death or injury is a deterrent to deliberate acts but not necessarily to accidental acts.

The applicant stated that, with regard to crediting the five valves, a configuration change would involve review by NCS, because NCS has a permanent position on the Plant Safety Review Committee (PSRC) that reviews all changes, not just changes to FMOs. The applicant stated that the example of the threat of imminent death or injury was intended to address situations that were obviously hazardous to life and health, as opposed to the broader spectrum of accidental actions. The examples cited were driving a forklift into a filled UF_6 cylinder or cutting into a UF_6 pipe. The NRC staff agreed that the examples were reasonable, given the additional details provided by the applicant, and that no further response was necessary beyond a written confirmation of USEC's statements made during the meeting as reflected above.

USEC Response

USEC confirms the statements made during the meeting as reflected above. The following clarifying information is provided. Quorum members of the PSRC are provided in plant procedures. Section 2.2.1 of the License Application describes the roles of the PSRC. In addition, the ACP change control process as it pertains to NCS is described in Section 5.3.3 of the License Application. This process ensures all proposed changes to the plant's configuration are appropriately reviewed by NCS personnel to ensure that there are no potential impacts to the NCS safety basis of the plant. No changes to the License Application are required.

8. NRC's RAI NC-17 requested the applicant to remove the statement from its application in Section 5.4.2.1 discussing controls used to preclude areas from being classified as fissile material operations. The applicant responded that controls relied on to prevent areas from meeting the criteria for being an FMO under abnormal conditions would not be double contingency controls or items relied on for safety (IROFS), but that such controls would be identified through a documented process.

   The NRC staff indicated that the status of such controls and how they are identified and maintained was not clear, and also that their inclusion in procedures or work instructions should not be left to the discretion of line managers.

   The applicant stated that controls to prevent exceeding the threshold for an FMO (enrichment greater than 1 wt% ^{235}U or greater than 100 g ^{235}U) would be identified as procedural controls, such as material control and accounting (MC&A) controls on cylinder movement credited with ensuring that only natural uranium is used initially in the feed facility. The applicant also stated that such controls would be uniquely identified in such procedures, in similar fashion to the way in which administrative NCS controls are stamped in operating procedures. The NRC staff agreed that this would be an acceptable way to prevent non-fissile operations from becoming FMOs, and that no further response was necessary beyond a written confirmation of USEC's statements made during the meeting as reflected above.

USEC Response

USEC confirms the statements made during the meeting as reflected above. However, the following clarifying information is provided. The only NCS controls that are required to be uniquely identified in procedures are those controls necessary to maintain the double contingency principle and/or IROFS. Handling of uranium contaminated items is controlled by procedures or work instruction to ensure appropriate safety measures are implemented for the protection of personnel and the public. Procedural controls required to maintain a non-fissile material operation below either the 100 g ^{235}U limit or the 1.0 wt. percent enrichment limit may be included in procedures or work instructions when needed to maintain the operation below these limits. Section 5.4.2.1 of the License Application has been clarified to state: "These controls can be either engineered or administrative and will be incorporated into applicable operating procedures or work

instructions when it is determined they are needed to maintain the non-fissile material operation below either 100 g ^{235}U or 1 wt. percent ^{235}U. This determination is made by the responsible line manager."

9. NRC's RAI NC-19 requested the applicant to clarify whether it would document exceptions to the preferred design philosophy in license application Section 5.4.3. The applicant responded that it did not intend to document the reasons for not following the preferred approach, such as when relying on administrative controls.

   The NRC staff indicated that, although it did not expect that engineered controls could be used in every instance, there should be a good reason for deviating from the preferred design philosophy and this reason should be documented.

   The applicant stated that administrative controls were used mainly in maintenance operations, in which use of passive or active engineered controls is impractical due to the hands-on nature of these operations. Examples of other operations involving administrative controls are the vacuuming of spills, container handling, and equipment transport. The requirement to ensure that criticality is highly unlikely under 10 CFR 70.61 drove the vast majority of ACP processes to rely on engineered controls. The NRC staff agreed that the design of the majority of the ACP facility relies extensively on inherently safe, passive equipment design, consistent with the applicant's commitment to the preferred design philosophy. As with other facility changes, replacement of an engineered control with an administrative control would require NRC pre-approval under 10 CFR 70.72. No further response was necessary beyond a written confirmation of USEC's statements made during the meeting as reflected above.

<u>USEC Response</u>

   Conformance with the preferred design approach is a natural result of compliance with the performance criteria of 10 CFR 70.61. The requirement to ensure that a criticality accident scenario is highly unlikely causes the vast majority of ACP operations to rely on passive engineered controls such as pipes, cylinders, cold traps, chemical traps, and other fixed equipment that maintains a pressure boundary for moderation control and, in some cases, a favorable geometry. Replacement of an engineered control with an administrative control will require NRC pre-approval pursuant to 10 CFR 70.72. USEC confirms the statements made during the meeting as reflected above. No changes to the License Application are required.

10. NRC's RAI NC-21 requested the applicant to remove the statement from license application Section 5.4.4 stating that exemptions from criticality accident alarm system (CAAS) coverage are documented in its NCS evaluations based on a conclusion that criticality is not credible. The applicant responded that the intent of the cited text was merely to point out that USEC would use form NCSEs to ensure that areas which do not have CAAS coverage will not contain greater than 700 g ^{235}U, and that the criteria immediately following were taken from the (700 g ^{235}U) mass limit in 10 CFR 70.24, the

(50 g $^{235}U/m^2$) areal density limit in ANSI/ANS-8.3, and the (5 g ^{235}U in any 10-liter volume) concentration limit in 10 CFR 71.53.

The NRC staff indicated that these two statements were inconsistent, in that the rule (10 CFR 70.24) does not allow for areal density or concentration based limits, but only contains a mass limit. Therefore, the use of areal density and concentration limits goes beyond what is allowed in the rule. Furthermore, the areal density criterion in ANSI/ANS-8.3 has not been endorsed by NRC as it is not in agreement with the rule; the concentration limit from 10 CFR 71.53 as applied to transportation packages has not been demonstrated applicable in a processing facility. Whereas mass is conserved by physical law, process conditions could result in consolidating or concentrating uranic materials under abnormal conditions.

The applicant stated that an example of where the areal density limit would be used would be a warehouse with contaminated equipment; an example of where the concentration limit would be used would be wastewater treatment. The applicant further stated that CAAS coverage would be provided if a credible concentration or accumulation mechanism existed, and that it would add the words "non-credible that those conditions change" to the words concerning the areal density and concentration criteria. The NRC staff stated that use of criteria beyond the 700 g ^{235}U limit from 10 CFR 70.24 requires an exemption request and should be included with the applicant's exemption request for the cylinder yards.

<u>USEC Response</u>

A change has been made to the first sentence in the third paragraph of Section 5.4.4 of the License Application to add "and that it is non-credible for these values to be exceeded." Also, Section 1.2.5 of the License Application has been revised to include a new Criticality Accident Alarm System exemption request for areal density and concentration criteria.

11. NRC's RAI NC-22 requested the applicant to provide technical justification for exemptions from CAAS coverage based on the criteria of less than 700 g ^{235}U, 50 g $^{235}U/m^2$, or 5 g ^{235}U in any 10-liter volume, and state whether this is applied to only normal or credible abnormal conditions as well. The applicant responded by referring to its response to question NC-21, and stated these criteria did apply to credible abnormal conditions.

The NRC staff indicated that the response to question NC-21 did not adequately justify use of the areal density or concentration criteria. As with question NC-21, an exemption request is needed to address the areal density and concentration criteria.

<u>USEC Response</u>

Section 1.2.5 of the License Application has been revised to include a new Criticality Accident Alarm System exemption request for areal density and concentration criteria.

12. NRC's RAI NC-23 requested the applicant to clarify whether all areas not covered by a CAAS exemption will be covered by two alarms, and also to state what compensatory measures would be used if dual alarm coverage is lost. The applicant responded that the CAAS system is required to provide dual coverage at all times, and that appropriate compensatory measures would be imposed until CAAS coverage was restored.

The NRC staff indicated that the response did not provide the requested clarification as to what compensatory measures would be used.

The applicant stated that the CAAS system was designed so that conditions in which CAAS coverage is lost are not anticipated and that appropriate facility-specific compensatory measures will be planned and documented as part of the design of the operation. The NRC staff stated that this would be adequate and no further response was needed beyond a written confirmation of USEC's statements made during the meeting as reflected above. The NRC staff indicated that it was not clear how or where the compensatory measures would be documented, but that it planned to review them as part of the ORR.

USEC Response

Compensatory actions for loss of CAAS coverage will be incorporated into off-normal operating procedures prior to initiation of operations that require CAAS coverage. No changes to the License Application are required.

13. NRC's RAI NC-24 requested the applicant to provide the technical basis for limiting the installation of evacuation horns and radiation warning lights to facilities within 200 feet of facilities with CAAS coverage. The applicant responded that the 200-foot evacuation zone is sufficient to ensure personnel are not exposed to doses exceeding regulatory limits.

The NRC staff indicated that the technical basis was not sufficient in that the analysis to show this fact was not provided. The applicant stated that this was the same criterion as currently used for the GDPs, and agreed to provide the technical basis for this criterion.

USEC Response

A calculation of the combined neutron and gamma dose from an initial burst of 10^{18} fissions at 200 feet using the method described in Regulatory Guide 3.34, subject to the 12 rad limit for excessive radiation dose from ANSI/ANS-8.3-1997, can be performed using the following formulas for gamma dose and neutron dose:

$$D_\gamma = \frac{(2.1\times10^{-20})N}{d^2 e^{3.4d}} \quad \text{and} \quad D_\eta = \frac{(7\times10^{-20})N}{d^2 e^{5.2d} Q}$$

Where N is the number of fissions, d is the distance from the source in kilometers, and Q is a quality factor of 3 for neutron dose. These formulas result in 4.593 rads and 4.573 rads, respectively, for a total dose of 9.166 rads, which is below the ANS/ANSI-8.3-1997 definition of excessive radiation dose of 12 rads. Similarly, using the calculation models in RG 3.34 for calculating prompt exposure from a criticality accident, a 10^{18} fission burst would result in an exposure of 4.593 rem and 13.719 rem for gamma and neutron radiation, respectively. The total of 18.313 rem for this estimated prompt exposure is below the performance requirements of 25 rem for high consequence events from 10 CFR 70.61(c)(1). This calculation conservatively assumes no shielding between the source and the recipient. Subsequent bursts at 10^{17} fissions will produce doses of 0.917 rads (1.831 rem) at 200 feet. Monitoring of the incident by emergency response forces will be performed to establish an evacuation zone that is adequate to maintain doses below regulatory limits for non-emergency responders. Emergency response dose control is described in the site Emergency Plan. This approach is consistent with the Technical Safety Requirements that apply to the adjacent GDP CAAS immediate evacuation zones at the Piketon site. No changes to the License Application are required.

14. NRC's RAI NC-29 requested the applicant to commit that process variables that can affect moderation will be identified as IROFS, when moderator control is relied upon. The applicant responded that moderator controls relied upon to ensure that criticality is highly unlikely would be identified as IROFS.

The NRC staff indicated that this response did not adequately address the question, which concerned process variables (e.g., temperature, pressure) were used to establish moderator control.

The applicant stated that, in the ACP, the main moderation control is the pressure boundary for both process equipment and UF_6 cylinders, and that all moderator controls in the facility are passive, and therefore the use of process variables does not apply.

The NRC staff stated that this is acceptable if moderator controls are all passive, and no further response was necessary beyond a written confirmation of USEC's statements made during the meeting as reflected above.

USEC Response

In UF_6 processing, moderation control of the UF_6 is provided by the pressure boundary of the equipment used to process the UF_6. The pressure boundary prevents moderation by moisture in ambient air, liquid water falling onto the equipment from external sources, or water from precipitation for outdoor equipment. The pressure boundary is a passive feature. If the analysis for an operation concludes that failure of the pressure boundary can credibly result in a criticality accident, then those passive features are IROFS. USEC confirms the statements made during the meeting as reflected above. No changes to the License Application are required.

15. NRC's RAI NC-32 requested the applicant to clarify what management measures will be used when relying on geometry control. The applicant responded that pre-operational verification walk-downs and pre-use verification would be used to ensure that geometry control for permanently installed equipment is implemented. In addition, permanently installed equipment may be subject to periodic surveillance if the equipment may credibly degrade.

The NRC staff indicated that the response should state that permanently installed equipment credited for geometry control will be subject to periodic surveillance if the equipment may credibly degrade.

The applicant stated that it will change "may" to "will" in the text cited above.

USEC Response

Permanently installed equipment used as a geometry control will have periodic surveillances performed if the equipment may credibly degrade. The specific surveillances performed and their frequency are determined through the configuration management program or are specific controls required by the NCSE. No changes to the License Application are required.

16. NRC's RAI NC-33 requested the applicant to clarify that, when parameters other than mass are relied on in setting mass limits, control of these parameters in conjunction with mass will only be credited as one leg of double contingency, and that if these items are not credited as IROFS, the associated parameter will be evaluated at its most reactive credible value. The applicant responded that highest credible or optimum moderation will be assumed when geometry, enrichment, or composition is used as the primary control.

The NRC staff indicated that this response only addresses moderation, and not other parameters, and does not address the underlying question which is whether both mass and the parameters used to determine safe mass could be credited as two distinct controls for meeting the DCP.

The applicant stated that, if mass limits are dependent on another controlled parameter, then this combination of controls will only be counted as one control for meeting the DCP. The NRC staff agreed that this is appropriate, and no further response was necessary beyond a written confirmation of USEC's statements made during the meeting as reflected above.

USEC Response

Section 5.4.5.1 of the License Application has been revised to include the statement: "When safe mass values are dependent on the geometry, enrichment, composition, or some other parameter, the combination of mass and the other parameter is used as one

control to meet the double contingency principle." USEC confirms the statements made during the meeting as reflected above.

17. NRC's RAI NC-39 requested the applicant to describe whether minimum reflector conditions are used to account for the presence of nearby structural or transient materials in facility calculations. The applicant responded that reflection conditions are described and justified in each NCSE, and that when operations involve the routine presence of personnel at the operating floor level, full water reflection or interstitial moderation combined with full density water blocks to simulate personnel may be used.

However, for operations in which the routine presence of personnel is not considered credible, full water reflection is not considered.

The NRC staff indicated that this response states that full water or interstitial reflection may be used, but does not state that one of these assumptions will be used, in evaluating the effect of personnel as reflectors. In addition, the response did not contain a justification for not having a minimum reflector condition to account for other transient reflectors.

The applicant stated that, in indicating that it may use option A (full water reflection) or B (interstitial moderation combined with full density water blocks to simulate personnel), the intent was not that it may use one of these options or some other option, but that it would commit to using either option A or B. The applicant stated that whatever reflection conditions were assumed would be justified to be bounding. The NRC staff questioned why the applicant did not commit to a minimum reflection condition for transient reflectors or structural materials that may be present in addition to operations personnel. The applicant stated that it did not consider this appropriate because there would be no solution processing in the facility, and that it would analyze any specific reflectors present in a given operation. The NRC staff agreed that this was acceptable, and no further response was necessary beyond a written confirmation of USEC's statements made during the meeting as reflected above.

USEC Response

NCS evaluations will include bounding reflection conditions for normal and credible abnormal conditions. USEC confirms the statements made during the meeting as reflected above. No changes to the License Application are required.

18. NRC's RAI NC-44 requested the applicant to describe the amount of margin used when deriving subcritical limits based on handbooks, and to describe the associated validation process. The applicant responded that handbooks are used as the starting point for establishing NCS limits, and that handbook limits may be used as written, incorporated into an operation-specific calculation, or applied in reduced fashion based on credible upset conditions. In addition, handbook data derived from experiment requires no validation, and handbook data derived from a calculational study is subject to the same validation requirements as other calculations.

The NRC staff indicated that this response appears to be inconsistent with that of question NC-31, which indicates that handbook limits are always supplemented by calculations. The response did not address the methods of validating handbook data that is used directly in lieu of operation-specific calculations, or how much margin of subcriticality is applied.

The applicant stated that the handbooks it used are based mostly on experimental data, rather than calculations, but that any handbooks used would be "vetted" prior to using them as the starting point for establishing NCS limits. The applicant agreed to provide an example of this.

<u>USEC Response</u>

Handbooks used for ACP operations are nationally recognized throughout the NCS industry as high quality analyses that have been confirmed through many years of use or based on experimental data. For example, LA-10860-MS, *Critical Dimensions of Systems Containing* ^{235}U, ^{239}Pu, *and* ^{233}U, 1986, is a handbook based on hundreds of critical experiments and provides an excellent source for obtaining values corresponding to critical configurations. Other handbooks are held to similar criteria for excellence, industry acceptance, and quality of data to be used at the ACP without further verification calculations. Because handbooks tend to give minimum critical or maximum subcritical values, use of these values for criticality controls is not appropriate to meet the double contingency principle. Instead, these values are reduced such that subcriticality can be demonstrated under normal and credible abnormal conditions. No changes to the License Application are required.

19. NRC's RAI NC-45 requested the applicant to describe how hand calculations are validated. The applicant responded that: (1) use of the modified two-group diffusion equation did not require validation because the method contains inherent conservatism and uses experimentally derived data; (2) use of buckling conversion is applicable to all types of material compositions and does not require validation; (3) comparative analysis refers to the use of handbook data is addressed in the response to question NC-44; (4) the use of the solid angle method is inherently very conservative and thus does not require validation; and (5) the use of the surface density method applies the reactivity of a single unit to an entire array, and that the single unit reactivity is determined using handbooks or validated methods, so that no specific validation is required.

The NRC staff indicated that the validity and adequate margin of subcriticality must be shown for any calculational method used to derive NCS limits. In addition, some of these methods have variants and therefore are not uniquely described in the license application. A detailed description of the methodology, the summary of systems or parameters over which it is valid, any assumptions or limitations that must be adhered to, and demonstration of adequate margin of subcriticality are all needed for such methods. In addition, if the method cannot be validated using critical benchmarks, a reference showing that it is an accepted and established method should be provided.

The applicant stated that it would provide a more detailed discussion of hand calculations and their use.

USEC Response

USEC does not intend to routinely use hand calculations to determine reactivity given the ready availability of computing resources. However, in the unlikely event that computing resources are unavailable and calculations are required, the hand calculations will be performed using methodologies described in "Nuclear Criticality Safety" by R.A. Kneif, American Nuclear Society, 1991 and subject to an upper safety limit of 0.95. Given the conservatisms inherent in the various methods, using an upper safety limit of 0.95 is adequate to ensure criticality safety. Section 5.4.5.2 of the License Application has been revised to include this commitment. The commitment states, "When hand calculations are used, the specific methodology employed will be as described in "Nuclear Criticality Safety" by R.A. Kneif, American Nuclear Society, 1991 and subject to a total system reactivity of 0.95 for all credible off-normal events."

20. USEC will provide any additional clarification or confirmation upon request by NRC.

21. USEC will provide any additional clarification or confirmation upon request by NRC.

22. NRC's RAI NC-50 requested the applicant to clarify to which of the currently NRC-endorsed ANSI/ANS-8 Series standards USEC is committing. The applicant responded that the details of what specific standards were being committed to are included in license application Section 1.4.

The NRC staff indicated that the response did not address whether the applicant committed to all the recommendations ("should" statements) or only the requirements ("shall" statements) of the affected standards, and did not justify why the applicant did not commit to certain of the ANSI/ANS-8 Series standards. The staff reviewed the set of commitments in license application Section 1.4 and concluded that the applicant had mainly committed to those ANSI/ANS-8 Series standards that were applicable to the ACP. The exception was ANSI/ANS-8.23; it was unclear whether the emergency response provisions of this standard are applicable to ACP operations.

The applicant stated that it would comply with all "shall" statements of the standards to which it was committing (except as qualified in license application Section 1.4), but that it would not necessarily comply with all "should" statements, with the exception of complying with the DCP in ANSI/ANS-8.1. The applicant further stated that a specific commitment to ANSI/ANS-8.23 was not necessary because its Emergency Plan compared favorably with the standard.

The NRC staff stated that it would review the Emergency Plan and discuss with the applicant later.

USEC Response

Section 2.2.4 of the Emergency Plan states that our emergency response to CAAS alarms and/or nuclear criticality events is consistent with guidance contained in ANSI/ANS 8.23-1997. Section 1.4 of the License Application has been revised to commit to this standard.

Enclosure 2 to AET 05-0038

Responses to Attachment 2 of the April 25, 2005 Meeting Summary
(Decommissioning Funding)

Enclosure 2 of AET 05-0038

1. In response to a request for additional detail in Table C.3.7 explaining the decontamination method to be utilized, USEC explained that it planned to use dry decontamination, including wiping, vacuuming, and light and heavy abrasion. Costs of those methods were reflected in Tables C.3.14 (waste components), C.3.15 (purchase of waste boxes), and C.3.17 (miscellaneous). USEC would perform little demolition, because it would use Department of Energy (DOE) buildings and be required to return the buildings to DOE. A small number of USEC-constructed buildings also were included in the cost estimate. Equipment would be removed and sent for disposal without decontamination whereas the centrifuge casings would be decontaminated. Building areas would be cleaned largely through vacuuming. Waste from vacuuming would be placed in waste box void areas. Vacuuming of floors was included in the estimate; cleaning of walls and ceilings was not included because the facility during operations would adhere strictly to ALARA and maintain a high level of cleanliness. The floors of all cylinder yards would be scarified, as needed, and waste from the scarification would also be disposed in the voids of the waste boxes.

   USEC was requested to provide additional detail on the unit costs to remove and package equipment, to scarify, and to vacuum (compared if possible to modeled representative costs in NUREG-6477).

<u>USEC Response</u>

In the Decommissioning Funding Plan (DFP) estimate, Table C3.7 entitled Decontamination or Dismantling of Radioactive Facility Components, the follow-on notes list the anticipated tasks considered in this estimate.

These tasks can be further delineated as follows for their respective labor efforts (person-days):

Administrative support	6,570
Engineering support	5,475
Security	18,615
Maintenance/Scheduling	1,095
Decontaminate Facilities (Internals Only)(Vacuuming); Waste Segregation; Erect Decontamination Facility (air structures/tenting)	2,190
Dismantle Centrifuge Machines	78,840
Dismantle Facilities/Components	21,900
Tails Disposition – Cylinder Movement/Disposition	6,570
Number Persons = 129	141,255

Decontamination of the facility internals will require 2,190 person-days of effort. The assumed preferred decontamination method employed during decommissioning is a dry method (i.e., vacuuming, wiping, dusting, scooping, sweeping, grinding, etc.), wherever practical, except for hard, stubborn areas that may need some special attention (i.e., dry – higher abrasion or wet - chemical cleaning). It is further assumed and anticipated that there will be a low amount of overall contamination due to a strict adherence to the As Low As Reasonably Achievable (ALARA) principles throughout the life of the plant.

The quantity of waste generated by employing dry decontamination methods is generally the lowest amount anticipated. Vacuuming the facility internals can reasonably accumulate approximately five B-25 boxes full of debris [Area = 1.04 x10^6 square feet (ft^2) (reference internal building floor area Table C3.5A), B-25 box = 90 cubic feet (ft^3), uniform thickness = 0.005 inches (in.)].

Additional details to decontaminate the facility internals are: the area is calculated to be 1.04 x 10^6 ft^2, the quantity of area decontaminated (vacuuming or dry/wet wiping) is assumed to be 750 ft^2/day, and there are two work crews anticipated. To dismantle the centrifuge machines, plans are to work 5 days per week, 24 hours per day utilizing 3 shifts. A shift is defined as 8 hours of work with a crew compliment of 24 persons. To dismantle/demolish the add-on facilities, the area is calculated to be 433,000 ft^2 (not including cylinder yards), the quantity of area dismantled is 333 ft^2/day, and the plans are to work five days per week, 16 hours per day utilizing two shifts. A crew compliment is 10 persons. The 2004 edition of RS Means was utilized to derive some of these unit costs.

Packaging of equipment is a detailed activity categorized under the dismantling of centrifuge machines as a labor activity, which is listed as an identified task in Table C3.7. Dismantling of centrifuge machines tasks considers these activities: machine removal and transport, machine disassembly, casing decontamination and storage, service module demolition, mezzanine piping demolition, handling and segregation of accumulated waste, health physics (HP) support, crew administration and supervision, and receipt and transport of B-25 waste containers/boxes.

In the DFP estimate, Table C3.8 entitled Restoration of Contaminated Areas on Facility Grounds, the follow-on notes list the anticipated tasks considered in this estimate. These tasks decontaminate the external facilities (i.e., cylinder yards). Again the assumption is a minimal amount of loose contamination. The labor estimate includes the HP survey, removal of fixed contamination (i.e., scarification of cylinder yards), and the collection and disposal of the waste. These tasks were estimated at 3,066 respective labor efforts (person-days).

The quantity of waste generated by scarifying the top 1/8-in. of the entire cylinder yard would accumulate approximately 184 B-25 boxes of debris [Area = 1,586,280 ft^2 (reference external cylinder yard area Table C3.5A), B-25 box = 90 ft^3, uniform thickness = 0.125 in.].

The debris accumulated from the internal vacuuming or the external scarifying dry decontamination efforts are mostly dust or granular in make-up and will be utilized as fill

material in the low-level, dry, unclassified waste disposal B-25 boxes generated from the crushed centrifuge machines components (19,200 B-25 boxes) described in Table C3.14 of the DFP.

In the DFP estimate, Table C3.14 entitled Packaging, Shipping, and Disposal of Radioactive Wastes, describes the waste streams anticipated as a result of decommissioning the American Centrifuge Plant (ACP). The waste streams are segregated into two distinct categories: Unclassified and Classified wastes. Centrifuge Machine wastes accumulate into these two categories as follows: low-level, contaminated, unclassified, liquid waste and low-level, contaminated, classified, solid waste. From the first table, there will be 295 55-gallon barrels of low-level, contaminated, unclassified, liquid waste accumulated, which will be shipped to Diversified Scientific Services Inc. in Oak Ridge, Tennessee for incineration. From the second table, there will be 19,200 B-25 boxes of low-level, contaminated, classified, solid waste accumulated for disposal, assumed to be at the Nevada Test Site. Furthermore, the other auxiliary equipment (from Table C3.5 of the DFP) is compacted into 665 B-25 boxes of low-level, contaminated, unclassified, solid waste accumulated for disposal at EnviroCare in Clive, Utah. A tertiary stream identified, but not totally defined is the storage of the centrifuge machine casings and service module structures. This equipment is decontaminated, but is set aside for later salvage though no salvage credit is taken for this material. This material is not considered waste and is not calculated in the waste stream.

Table C3.14 of the DFP built into the waste disposal unit costs is the equipment and labor needed to obtain loaded and staged B-25 boxes from the machine dismantling process for further on-site disposition (i.e., survey, inspection, sealing, handling, labeling, etc.) and staging for ultimate transport to the various waste disposal facilities identified in Table C3.14 of the DFP.

Table A-2, Appendix A of NUREG/CR-6477 lists a few surface decontamination rates for vacuuming, wiping, and concrete scabbling activities. A comparison of these unit rates to those made in USEC's DFP was made. The values were gleaned from the DFP estimate, with appropriate unit conversions.

Surface Rates	**NUREG Rate (m^2/h)**	**USEC Rate (m^2/h)**	**Comments**
Dry Vacuuming	60	8.7	DFP Table C3.8 – 750 ft^2/d
Dry/Wet Wiping	30	8.7	DFP does not distinguish between vacuuming and wiping
Concrete Scabbling	10	21 or 17	DFP Table C3.7 details as described in the response given above; Area = 1,586,280 ft^2, Dur = 438 d; 2 crews; depends upon hours worked per day (8 or 10 hours respectively)

The table comparison analysis shows that USEC's DFP vacuuming and dry/wet wiping activities are quite low (i.e., less area is covered per unit time). USEC's DFP estimate was determined utilizing other experience gained from Portsmouth Gaseous Diffusion Plant to calculate the manpower effort and appears to be very conservative. The concrete scabbling rate calculated for USEC's DFP is somewhat quicker than the NUREG rate, but reasonable based on the activity performed. The differences are likely attributable to the type of area being addressed. The NUREG has more obstacles (barriers, start/stop operation that tend to reduce the working rate) for the listed building demolition effort, whereas USEC's scabbling effort in on wide open cylinder yards with little or no obstacles.

Another common table layout utilized throughout NUREG/CR-6477 can be found in Section 6.2 Decommission Analysis on page 6.3. To help simplify a comparison between this NUREG and USEC's DFP estimate, USEC has developed a similar layout as depicted in the NUREG document with comparable units and is provided below:

Summary of Estimated Values for Decommissioning the ACP Commercial Plant to NRC "Free-Release" Criteria

Parameter	Planning and Preparation	Decontamination and Dismantling	Ground Restoration	Final Rad Surveys	Site Characterization and Long Term Surveillance	Total
Duration (days)	219	1,095	438	548	1,314	1,314
Effort (pers-days)	7,884	141,255	3,066	3,285	6,570	162,060
Manpower (persons)	36	129	7	6	5	183
Occupational dose (pers-rem)	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
Cost ($ millions)						
Staff	2.578	39.591	0.738	0.963	2.357	46.227
Packaging/Shipping/Waste Disposal		47.511				47.511
Equipment/Supply Costs		14.974				14.974
Laboratory Costs		1.353				1.353
Miscellaneous Material Costs		2.500				2.5
NRC Staff Review and Approval		0.08				0.08
NRC Fees		18.6				18.6
DOE Lease		6.0				6
Business Insurance		0.300				0.3
Taxes		0.181				0.181
Indirect Services		33.581				33.581
Subtotals	2.578	164.671	0.738	0.963	2.357	171.307
G&A	0.155	9.88	0.044	0.058	0.141	10.278
15% Contractor Profitability	0.410	22.481	0.117	0.153	0.375	23.536
25% Contingency	0.786	49.258	0.225	0.294	0.718	51.280
Totals	3.929	246.290	1.124	1.468	3.591	256.401

- Decontamination Duration - six years
- Centrifuge machine internals disposed as "classified, contaminated, low-level" waste
- Centrifuge casings and service module structural steel disassembled, decontaminated, and stored for later disposition
- Most centrifuge auxiliaries disposed as "unclassified, contaminated, low-level" waste
- Decontamination efforts are performed on all "horizontal" surfaces
- Compaction of waste volume utilized
- No salvage
- 219 days is representative of productive effort per year

[Centrifuge and centrifuge-related equipment is disposed; all other equipment and buildings/yards added by USEC remain, but decontaminated to the NRC "free-release" criteria]

Again, the values utilized were obtained directly from the DFP estimate and placed into the comparable layout. The table comparison analysis for USEC's DFP estimate is quite conservative.

2. In response to a request for additional detail on the labor overhead rate, USEC explained that proprietary information had been provided that contained the requested information. The NRC informed USEC that ICF Consulting would review this information and USEC would be contacted if any further clarifications were needed.

   NRC to review information associated with this item and provide USEC any additional need for clarification/confirmation.

USEC Response

As discussed with NRC and ICF Consulting, USEC believes we have adequately addressed the overhead allocation issues as noted on the tables. These allocations are representative of the costs that would be associated with doing business on the DOE reservation in Piketon for either USEC or a third party contractor.

3. In response to a request for additional detail on how USEC had derived the two consolidated labor categories (i.e., salary and hourly) from the eleven labor groupings stated to have been included in the cost estimate, USEC explained that proprietary information had been provided that contained the requested information. The NRC informed USEC that ICF Consulting would review this information and USEC would be contacted if any further clarifications were needed.

   NRC to review information associated with this item and provide USEC any additional need for clarification/confirmation.

USEC Response

As a result of discussions with NRC and ICF Consulting, USEC agreed that it was reasonable to include wage escalation in its labor estimate. Based on past experience for wage escalation relevant to non-union employees at the Portsmouth Gaseous Diffusion Plant, USEC will escalate wages at a 3 percent rate in years two through six of the estimate. These updated tables will be submitted to the NRC under separate cover (AET 05-0040).

4. In response to a request for additional detail on waste disposal costs for any wastes generated by decontamination of the facility, USEC explained that such wastes would include primarily waste from vacuuming floors, waste from scarifying floors, and wastes from wipes used to sample for contamination. The estimated amounts of such wastes were well within the container limits of the waste boxes, and the small articles would occupy free volume in those boxes. USEC agreed to add a statement to the decommissioning cost estimate (DCE) stating those assumptions.

USEC Response

In the DFP estimate, Table C3.14, a footnote was added in Revision 1 of the DFP and can be found under the second table, "Assumptions," which states, "B-25 boxes contain volume gaps, which are filled to capacity from scarified yard materials/debris." Further explanation on waste streams is documented in USEC's response to NRC Summary # 1 above.

5. In response to a request for additional justification for the estimated costs of packaging, shipping, and disposing of tails expressed in dollars per kgU, USEC explained the following: The estimate of the contract price with the Department of Energy was in 1998 dollars, which USEC believes is still accurate. USEC was aware of a higher estimate given to Louisiana Energy Services (LES) by DOE for tails disposal. However, USEC had validated the cost per kgU that it was using, based on a proprietary bid that it had submitted to DOE for the contract to build the Uranium Disposition Services (UDS) facility for tails conversion. USEC's estimate was based on: (1) cost information from its bid; (2) cost information from the UDS contract; (3) the amount that USEC would be willing to pay the DOE, taking into consideration that USEC could build its own facility; and (4) the fact that USEC would not have transportation costs because of the proximity of its proposed ACP to the UDS conversion facility.

USEC Response

USEC's response is considered USEC Proprietary Information pursuant to 10 CFR 2.390 and is being submitted under separate cover (AET 05-0037).

6. In response to a request for additional justification for the use of a 10 percent rather than a 25 percent contingency factor on the cost of tails conversion, USEC explained that it believed the smaller contingency was justified because: (1) it was committing to prepare annual revisions rather than revisions every three years to the DCE following full installation of all centrifuges in 2010; and (2) the amount of tails generated is a consistent function of the number of machines and can be determined by a simple mass balance calculation. USEC confirmed that it did not intend to use an inflation adjustment to recalculate its DCE, but rather in all cases to recalculate based on the estimated costs of the activities.

USEC Response

USEC's response is considered USEC Proprietary Information pursuant to 10 CFR 2.390 and is being submitted under separate cover (AET 05-0037).

7. In response to a request for additional justification for the absence of sampling costs (except urinalysis), USEC explained that it did not expect to perform any sampling, such as soil analysis or core sampling that would require analysis by an off-site laboratory. Sampling costs were captured in Table C.3.9 for labor for surveys, wipe tests, and on-site analysis of wipes, and in Table C.3.17 for supplies. The storage yard would be surveyed and then scabbled, with the process repeated as many times as necessary. USEC stated that this

approach was based on experience with the GDP yards, but that the ACP yards were expected to be newer and cleaner.

USEC Response

A footnote has been added to Tables C3.8 entitled Restoration of Contaminated Areas of Facility Grounds and C3.9 entitled Final Radiation Survey, of the DFP under "Assumptions," which states, "Labor estimate includes non-labor costs for analytical sampling/surveying efforts."

USEC does not anticipate obtaining any samples (i.e., core sampling or soil analysis, etc.) that would require an off-site laboratory analysis for the following reasons: 1) USEC leases the buildings from the DOE and are obligated to return the majority of the buildings back to DOE. Those buildings not identified in the lease are captured in the estimate to be totally demolished, 2) there are known DOE ground legacy issues, where USEC is not liable, 3) the concrete is of sufficient hardness and density per design specifications that core sampling is not the primary means to detect absorbed contamination, and 4) instead of sampling, many survey methods will be employed to detect contaminated areas. One such example is the process anticipated to be utilized in the cylinder storage yards. All USEC listed cylinder yards will be surveyed. If a contaminated area is located, it will be marked and decontaminated later. The anticipated methods of decontamination in the cylinder yards are scarify/scabble then vacuum with the survey/decontamination process repeated as often as necessary to qualify the area clean (Free Release criteria). The decontamination debris will be disposed of in B-25 boxes as fill material (see USEC's response to NRC Summary # 1 above). This method has been utilized successfully, with experience gained from the GDP yards. In addition, the ACP yards will be newer and will have lower contamination threshold identification limits applied to maintain the yards in accordance with the ALARA principle.

Enclosure 3 to AET 05-0038

Changed Pages for the License Application and Supporting Documents

Remove and Insert Instructions
Revision for the American Centrifuge Plant

Remove and Properly Destroy	**Insert**
LA-3605-0001, License Application	
Cover Page – Revision 2	Cover Page – Revision 3
Inside Cover Page – Revision 2	Inside Cover Page – Revision 3
ULOEP-1/ULOEP-2	ULOEP-1/ULOEP-2
1-19/1-20 1-47/1-48 1-51 through 1-56 1-101 through 1-106 1-113 through 1-116	1-19/1-20 1-47/1-48 1-51 through 1-56 1-101 through 1-106 1-113 through 1-116
5-5 through 5-18	5-5 through 5-18
10-5/10-6 10-9/10-10 10-17/10-18 10-21/10-22	10-5/10-6 10-9/10-10 10-17/10-18 10-21/10-22
NR-3605-0006, Decommissioning Funding Plan	
Cover Page – Revision 1	Cover Page – Revision 2
Inside Cover Page – Revision 1	Inside Cover Page – Revision 2
ULOEP-1/ULOEP-2	ULOEP-1/ULOEP-2
C-11through C-14 C-21/C-22	C-11through C-14 C-21/C-22

LA-3605-0001

License Application

for the American Centrifuge Plant
in Piketon, Ohio

The American Centrifuge™

USEC's Advanced Uranium Enrichment Technology

Revision 3

Docket No. 70-7004 May 2005

Information contained within
does not contain
Export Controlled Information

Reviewer: Original signed by RL Coriell
Date: 05/23/05

LA-3605-0001

LICENSE APPLICATION

for the American Centrifuge Plant
in Piketon, Ohio

Docket No. 70-7004

Revision 3

Information contained within
does not contain
Export Controlled Information

Reviewer: Original signed by RL Coriell
Date: 05/23/05

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UPDATED LIST OF EFFECTIVE PAGES

Revision 0 – 10 CFR 1045 review completed by L. Sparks on 07/29/04 and the Export Controlled Information review completed by R. Coriell on 07/30/04.
Revision 1 – 10 CFR 1045 review completed by L. Sparks on 03/04/05 and the Export Controlled Information review completed by R. Coriell on 03/10/05.
Revision 2 – 10 CFR 1045 review completed by J. Weidner on 04/29/05 and the Export Controlled Information review completed by R. Coriell on 04/29/05.
Revision 3 – 10 CFR 1045 review completed by J. Weidner on 05/23/05 and the Export Controlled Information review completed by R. Coriell on 05/23/05.

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5-18	3	11-2	1
6-5	1	11-9	1
6-7	1	11-18	2
6-9	1	11-37	1
7-8	1		
7-12	1		
7-13	2		
7-15	2		
7-18	1		
9-6	1		
9-7	1		
9-8	1		
9-9	1		
9-10	1		
9-11	1		
9-12	1		
9-14	1		
9-15	1		
9-16	1		
9-17	1		
9-19	1		
9-21	1		
9-23	1		
9-25	1		
10-2	1		
10-5	3		

The major components that support the withdrawal operations are withdrawal (compression) trains, cold boxes, cold traps, assay spectrometers, and vents.

1.1.5.5.5 Sampling and Transfer Operations

UF_6 sampling and transfer operations for UF_6 product material is carried out in the product operations area of the X-3346 building. Since the American Society for Testing and Materials (ASTM) sampling standards necessitate that sampling must be from homogenized UF_6, the design involves liquid UF_6 material in the source cylinders and the transfer operations. Autoclaves with heating and cooling capability are used to liquefy UF_6 in the source cylinders, to facilitate transfer into customer cylinders and then to solidify the UF_6 heel remaining in the cylinders at the end of the transfer. The autoclaves are pressure vessels and are designed to contain a UF_6 release. Electrically heated hot air is the heating medium and cold air is used for cooling.

The major components that comprise the sampling and transfer operations are autoclaves, cold traps, and vents.

1.1.5.5.6 Receiving and Shipping Operations

The X-3346A building is the usual receiving point for cylinders. Full feed cylinders (14-ton), customer cylinders (2.5-ton), tails cylinders (14-ton), and overpacks with customer cylinders are received and weighed in the X-3346A building, then the cylinders and overpacks are transported to storage until needed.

The X-3346A building is also the shipping point for emptied cylinders leaving the ACP as well as UF_6 cylinders shipped to fulfill customer product orders, although any approved UF_6 cylinder may be shipped from this facility. See Figure 1.1-4 (located in Appendix B) for a schematic of the Feed, Withdrawal, and Product Operations.

Filled customer product cylinders, emptied feed cylinders, and other UF_6 cylinders will be prepared for shipment and shipped in accordance with U.S. Nuclear Regulatory Commission (NRC) and DOT regulatory requirements from the X-3346A.

1.1.5.5.7 Waste Handling Operations

Depleted UF_6 tails material is considered a resource material with the ultimate disposition to be determined and is not considered a waste. USEC intends to evaluate possible commercial uses for depleted UF_6. Depleted UF_6 is stored in steel cylinders within cylinder storage yards until this material can be processed in accordance with the disposition strategy established by USEC. Depending upon technological developments and the existence of facilities available prior to the ACP shutdown, the depleted UF_6 may have commercial value and may be marketable for further enrichment or other processes.

Waste generated by the ACP is collected, handled, packaged, segregated, stored, and shipped for off-site treatment/disposal in a safe and environmentally acceptable manner in

accordance with applicable state and federal regulations, and plant procedures. Waste accumulation areas are established throughout the ACP as necessary to meet these regulatory requirements.

The ACP obtains waste management services from a qualified provider licensed/certified by the NRC or an agreement state. Waste may be further sampled/measured to assist in determining the proper waste characterization and proper disposal/treatment method.

Potential waste streams generated include Low-Level Radioactive Waste, LLMW, RCRA Hazardous Waste, Sanitary/Industrial Waste, Recyclable Waste, and Classified/Sensitive Waste.

Waste generating activities are evaluated for waste minimization opportunities to reduce the volume and toxicity of waste generated to the degree determined to be economically practicable.

A further description of the transportation impacts can be found in Section 4.2 and the waste impacts can be found in Section 4.13 of the Environmental Report for the American Centrifuge Plant.

1.1.5.5.8 Liquid and Air Waste Discharge Points

Waste discharge points are categorized by either liquid (water) or air.

For liquid, wastewater discharges are handled by different means depending upon the originating source: process, sanitary, or storm water.

No process wastewater is intentionally discharged from the liquid effluent tanks. Accumulated water in these tanks are sampled and managed according to analytical results. Trained professionals using approved spill response protocols and spill response equipment will promptly contain liquid spills within the process buildings. Spill materials will be collected, sampled, analyzed, and managed in accordance with applicable federal and state laws. The only intentional process wastewater discharge resulting from plant operations is the blow down from the TWC (Tower Cooling Water) system. This cooling water system is not interconnected with the MCW (Machine Cooling Water) system located in the process buildings. The MCW system is a closed-loop system, which requires minimal makeup water, but does not have blow down discharges.

Sanitary wastewater (e.g., showers, toilets, etc.) located within the area discharge to the plant sanitary sewer system and ultimately to the X-6619 Sewage Treatment Plant. Treated sanitary wastewaters are discharged from X-6619 directly to the Scioto River via an underground pipeline via a permitted NPDES outfall.

Storm water runoff from the ACP area, along with some once-through cooling water (sanitary water), drain to a pair of holding ponds (X-2230N West Holding Pond and X-2230M Southwest Holding Pond). These ponds provide a quiescent zone for settling suspended solids, dissipation of chlorine, and oil diversion and containment. The ponds discharge to unnamed

1.2 Institutional Information

USEC Inc. is the applicant for the ACP license.

1.2.1 Corporate Identity

USEC is a global energy company and its subsidiary, the United States Enrichment Corporation, is the world's leading supplier of enriched uranium fuel for commercial nuclear power plants. USEC, including its wholly owned subsidiaries, was organized under Delaware law in connection with the privatization of the United States Enrichment Corporation.

USEC is responsible for the design, manufacturing, assembling, installation, operation, maintenance, modification, and testing of the ACP in Piketon, Ohio.

USEC's principal office is located at 6903 Rockledge Drive, Bethesda, MD 20817. USEC is listed on the New York Stock Exchange under the ticker symbol USU. Private and institutional investors own the outstanding shares of USEC. The principal officers of USEC are listed below and are citizens of the United States.

James R. Mellor, President and Chief Executive Officer
Lisa E. Gordon-Hagerty, Executive Vice President and Chief Operating Officer

Philip G. Sewell, Senior Vice President
Robert Van Namen, Senior Vice President
Ellen C. Wolf, Senior Vice President and Chief Financial Officer
W. Lance Wright, Senior Vice President
James F. McDonnell, Vice President, Chief Information and Security Officer

The mailing address for the ACP is:

USEC Inc.
American Centrifuge Plant
P. O. Box 628
Piketon, Ohio 45661

The NRC has issued Certificates of Compliance to the United States Enrichment Corporation, a wholly owned subsidiary of USEC, to operate the Paducah and Portsmouth GDPs (Docket Numbers 70-7001 and 70-7002, respectively). Consistent with the requirements in 10 CFR 76.22 and in connection with the issuance of these Certificates, the NRC has determined that USEC is neither owned, controlled, nor dominated by an alien, a foreign corporation, or a foreign government. Issuance of a license to USEC would be consistent with the requirements of 10 CFR 40.38 and 70.40, since the NRC concluded that USEC has satisfied similar requirements in 10 CFR 76.22. Furthermore, more recently the NRC has issued a license to USEC to operate the Lead Cascade Demonstration Facility (Docket No. 70-7003) pursuant to 10 CFR Part 70. There have been no changes in ownership or control that would invalidate the NRC's previous findings.

Further, issuance of a license would not be inimical to the common defense and security of the United States or to the maintenance of a reliable and economical domestic source of enrichment services. To the contrary, issuance will support those important goals. Commercial deployment of American Centrifuge technology by USEC will help ensure the United States will continue to maintain a reliable and economic, domestic source of enriched uranium. Deployment of the ACP is in furtherance of the goals of the June 17, 2002, DOE-USEC Agreement to "facilitate the deployment of new, cost effective advanced enrichment technology in the United States on a rapid schedule." It will enable USEC to deploy a modern, efficient and reliable enrichment plant to supplement and replace its current 50+ year-old GDPs.

1.2.1.1 Site Location

The ACP is located on the DOE reservation. The reservation is located at latitude 39°00'30" north and longitude 83°00'00" west measured at the center of the reservation on approximately 3,700-acres of federally owned land in Pike County, Ohio, one of the state's lesser populated counties. The largest cities within an approximate 50-mile radius are Portsmouth, Ohio, located approximately 27 miles to the south, and Chillicothe, Ohio, located approximately 27 miles to the north. The reservation occupies approximately 750 security-fenced acres and is located about one and one half miles east of U.S. Route 23 and two miles south of U.S. Route 32, and two miles east of the Scioto River.

USEC, through its subsidiary the United States Enrichment Corporation, leases a significant portion of the DOE reservation from the DOE. The ACP is within the space leased by the United States Enrichment Corporation and occupies approximately 200 acres of the southwest quadrant of the CAA. USEC and its agents will conduct USEC activities within the ACP buildings/facilities and access and egress thereto, in accordance with this license application.

1.2.1.2 Other Reservation Activities

The United States Enrichment Corporation operates the GDP in accordance with a NRC Certificate of Compliance issued pursuant to 10 CFR Part 76 requirements. These operations include:

- Maintaining the GDP in Cold Standby status under a contract with the DOE;
- Performing uranium deposit removal activities in the cascade facilities; and
- Removing technetium-99 (^{99}Tc) from potentially contaminated uranium feed in accordance with the June 17, 2002 agreement between DOE and the United States Enrichment Corporation.

The United States Enrichment Corporation also possesses a license for radioactive material operations from the State of Ohio for the conduct of laboratory and associated support activities. This license encompasses laboratory analyses, in-field analyses for radioactive material deposits, health physics survey, and characterization activities.

1.2.3 Type, Quantity, and Form of Licensed Material

The type, quantity, and form of NRC-regulated special nuclear, source, and by-product material are shown in Table 1.2-1.

1.2.4 Authorized Uses

The ACP enriches UF_6 up to 10 wt. percent ^{235}U. The specific authorized uses for each class of NRC-regulated material are shown in Table 1.2-2.

USEC will provide a minimum 60-day notice to the NRC prior to initial customer product withdrawal of licensed material exceeding 5 wt. percent ^{235}U enrichment. This notice will identify the necessary equipment and operational changes to support customer product withdrawal, storage, processing, and shipment for these assays.

1.2.5 Special Exemptions or Special Authorizations

The following exemption to the applicable 10 CFR Part 20 requirements are identified in Section 4.8 of this license application:

- UF_6 feed, product, and depleted uranium cylinders, which are routinely transported inside the DOE reservation boundary between ACP locations and/or storage areas at the ACP, are readily identifiable due to their size and unique construction, and are not routinely labeled as radioactive material. Qualified radiological workers attend UF_6 cylinders during movement.

- Containers located in Restricted Areas within the ACP are exempt from container labeling requirements of 10 CFR 20.1904, as it is deemed impractical to label each and every container. In such areas, one sign stating that every container may contain radioactive material will be posted. By procedure, when containers are to be removed from contaminated or potentially contaminated areas, a survey is performed to ensure that contamination is not spread around the reservation.

- In lieu of the requirements of 10 CFR 20.1601(a), each High Radiation Area with a radiation reading greater than 0.1 roentgen equivalent man per hour (rem/hour) at 30-centimeters (cm) but less than 1 rem/hour at 30 cm is posted Caution, High Radiation Area and entrance into the area shall be controlled by an RWP. Physical and administrative controls to prevent inadvertent or unauthorized access to High and Very High Radiation Areas are maintained.

The on-site radiological impacts from the proposed exemptions to the requirements of 10 CFR 20.1904 and 20.1601 would be minimal and are consistent with previously approved exemptions found in the GDP certification. Moreover, pursuant to the regulations in 10 CFR 20.2301, the requested exemption is authorized by law and would not result in undue hazard to life or property.

The following exemption from the applicable 10 CFR 70.50 reporting requirement is identified in Section 11.6.3 of this license application:

- The 10 CFR 70.50(c)(2) reporting criteria require that the ACP submit a written follow-up report within 30 days of the initial report required by 10 CFR 70.50 (a) or (b) or by 10 CFR 70.74 and Appendix A of Part 70. In lieu of the 30-day requirement described in 10 CFR 70.50(c)(2), NRC approval to submit the required written reports within 60 days of the initial notifications is hereby requested.

  10 CFR 70.17 allows the Commission, upon application of any interested person or upon its own initiative, to grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The requested exemption is authorized by law because there is no statutory prohibition on extending the reporting period to 60 days.

  Furthermore, granting this exemption request will not endanger life or property or the common defense and security, in that the exemption request does not relieve the ACP from other requirements contained in 10 CFR 70.50 (a) or (b) or by 10 CFR 70.74 and Appendix A of Part 70, such as 1-hour, 4-hour, and 24-hour reporting requirements for defined events.

  The proposed exemption would result only in written reports being submitted within the time limit currently allowed under 10 CFR 50.73 for commercial nuclear power plants. It would be consistent with the exemption granted to the gaseous diffusion plants for reporting of events pursuant to 10 CFR 76.120(d)(2) (67 Federal Register 68699, November 12, 2002) and the exemption granted to the Lead Cascade during licensing.

  This proposal allows for completion of required root cause analyses after event discovery and fewer supplemental reports, thereby reducing regulatory burden and confusion. Thus, it is clearly consistent with the public interest.

  USEC notes that the requirements of 10 CFR 20.2201 and 20.2203 require written reports of certain events within 30 days after their occurrence. USEC is not requesting an exemption from these reporting requirements.

The following exemption from the requirements of 10 CFR 70.25(e) addressing the decommissioning funding requirements is identified in Section 10.10.4 and the Decommissioning Funding Plan (DFP) of this license application:

- 10 CFR 70.25(e) requires, in part, that "The decommissioning funding plan must also contain a certification by the licensee that financial assurance for decommissioning has been provided in the amount of the cost estimate for decommissioning...". As noted in Section 10.3 of this license application, the financial assurance for decommissioning the plant, to include the disposition of the UF_6 tails, which constitutes a major portion of the decommissioning liability, will be provide

incrementally as centrifuges are installed, operated on process gas, and UF_6 tails generated. In this way, funds will be made available as the decommissioning liability is incurred.

This exemption is justified for the following reasons: 1) It is authorized by law because there is no statutory prohibition on incremental funding of decommissioning costs. 2) The requested exemption will not endanger life or property or the common defense and security for the following reasons: the unique modular aspects of the American Centrifuge technology allow enrichment operations to begin well before the full capacity of the plant is reached. Thus, the decommissioning liability is incurred incrementally as more centrifuge machines are added to the process, until full capacity of the facility is reached; at which point the UF_6 tails are generated at a relatively constant rate throughout the life of the plant. As such, requiring full funding for decommissioning liability, to include UF_6 tails disposition, incurred over the lifetime of the plant, at the time of initial license issuance, produces an unnecessary financial burden on the licensee.

Furthermore, incremental funding of decommissioning costs, to include UF_6 tails disposition, is justified based upon USEC's commitments to update the cost estimates and provide a revised funding instrument for decommissioning and UF_6 tails disposition prior to operation of each additional increment of capacity on process gas, and after full capacity has been reached to annually adjust the cost estimate for UF_6 tails disposition and to adjust all other decommissioning costs periodically, and no less frequently than every three years. In addition, the relative stability of the factors, which are utilized to generate the UF_6 tails volumes, allows actual inventory values to be provided for prior periods of operation and reliable estimates for the upcoming periods of operation. The NRC has previously accepted an incremental approach to decommissioning funding costs for the United States Enrichment Corporation's operation of the GDPs. 3) Finally, granting this exemption is in the public interest for the same reasons as stated above and will facilitate deployment of gas centrifuge enrichment technology by eliminating an unnecessary financial burden on the licensee.

The following exemptions from the requirements of 10 CFR 70.24 addressing criticality monitoring are identified in Section 3.10.6 of the ISA Summary and discussed in Section 5.4.4 of this License Application. Exemptions are required for criticality monitoring of the UF_6 cylinder storage yards and from the 700 g ^{235}U limit where ^{235}U areal density or concentration levels do not exceed specified values.

- 10 CFR 70.24, *Criticality Accident Requirements*, requires that licensees authorized to possess special nuclear material in a quantity exceeding 700 g of contained ^{235}U shall maintain in each area in which such licensed special nuclear material is handled, used, or stored, a monitoring system capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of two meters from the reacting material within one minute.

10 CFR 70.17 allows the Commission, upon application of any interested person or upon its own initiative, to grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The requested exemption is authorized by law because there is no statutory provision prohibiting the grant of the exemption. The requested exemption will not endanger life or property or the common defense and security and is otherwise in the public interest for the reasons discussed below.

Transportation, handling and storage of solid UF_6 filled cylinders are doubly contingent. Double contingency is established by multiple controls that limit the likelihood for a solid product cylinder to be breached during transportation, handling or storage, and the likelihood for a breach to not be identified and repaired before sufficient moderation results in a criticality. Moderation control of UF_6 filled cylinders is maintained by ensuring cylinder integrity through periodic cylinder inspections. If a UF_6 filled cylinder is found to be breached, the cylinder is covered within 24-hours after discovery to reduce the potential accumulation of moderating material, i.e., rainwater. This time limit ensures a corresponding heavy rainfall will not result in accumulation of sufficient amounts of water to cause a criticality. Damaged cylinders are repaired as necessary and emptied. UF_6 cylinders are uniquely identified and their design requirements are controlled to further ensure cylinder integrity and reliability (i.e., UF_6 cylinders are QL-1 components and are controlled in accordance with the Quality Assurance Program Description), and USEC implements onsite cylinder handling practices (i.e., requiring the use of approved equipment in accordance with approved procedures), which reduces the likelihood that a solid UF_6 cylinder would be breached. These requirements are established as items relied on for safety to ensure the health and safety of the public and workers.

The UF_6 cylinders stored in storage yards are not covered by a criticality monitoring system unless those cylinders contain licensed material greater than 5.0 weight percent ^{235}U. NCS evaluation of product cylinders of any size, configured in infinite planar arrays, containing material enriched up to 5.25 weight percent ^{235}U, has concluded that subcritical conditions are maintained. The ACP ISA has concluded that cylinders containing licensed material less than or equal to 5.0 weight percent ^{235}U cannot be involved in a criticality accident sequence that has a probability of occurrence that exceeds 5×10^{-6}/year.

The frequency of criticality events in the cylinder yards have been decreased to the Highly Unlikely range ($<10^{-5}$/year) through the establishment of preventive controls established by the ISA in accordance 10 CFR 70.62. Considering the conservatism of the ISA methodology in developing the unmitigated frequency and actual historical data related to cylinder operations, the frequency values could be reduced further. This additional reduction considers the fact that during 50 years of GDP operations, only one cylinder breach has occurred due to

mishandling or equipment failure. Since that occurrence, cylinder handling equipment has been redesigned and cylinder handling methods have been revised to minimize the potential for breaches to occur. Another fact not considered in the ISA is that holes with a dimension of less than one inch will self-seal such that moderating material cannot infiltrate the breach. A third factor not considered in the ISA is that enriched cylinder operations require constant use and monitoring of cylinders such that corrosion breaches in enriched cylinders are highly unlikely. Allowing for this additional reduction in frequency, the probability for a criticality event becomes incredible, therefore CAAS coverage is not necessary.

The increased vehicular and pedestrian traffic in support of CAAS maintenance and calibration requirements would cause a subsequent increased likelihood for impact events involving cylinders and there would be an increased safety risk for workers from radiation exposure due to the ongoing CAAS maintenance and calibration requirements. To meeting the CAAS coverage requirements in ANSI 8.3 and the operating requirements for the ACP, enriched cylinder storage yards would require a minimum of 60 clusters. Clusters would need to be at a height of approximately 40 feet, which would require maintenance equipment and pedestrian traffic to perform testing and preventative maintenance tasks to ensure their reliability and operability. This equipment and traffic would increase the likelihood for fire and impact events in the cylinder storage yards such that workers would be at a higher risk for injury and exposure relative to the minimal mitigative value produced by the presence of CAAS.

- The ACP may operate storage areas that include more than 700 grams ^{235}U, but are limited to an areal density of 50 grams ^{235}U per square meter or a concentration of 5 grams ^{235}U in any 10 liter volume. When established through an approved Nuclear Criticality Safety Evaluation that either the areal density limit or the concentration limit is met for all normal and credible abnormal conditions, USEC is not required to maintain a criticality accident alarm system for those areas. This exemption is consistent with the language in 10 CFR 70.24 that refers to 700 grams of *contained* ^{235}U [emphasis added]. Typical storage containers are 55 gallon drums or B-25 boxes. Neither of those containers can contain more than 700 grams ^{235}U at the areal density or concentration limit listed above.

  ANSI/ANS 8.3-1997, *Criticality Accident Alarm System*, Section 4.2.1 does not require areas with less than 50 grams ^{235}U per square meter to have a criticality accident alarm system. 10 CFR 71.53(3) *Packaging and Transport of Radioactive Material* exempts fissile materials containing less than 5 grams ^{235}U in any 10 liter volume from compliance with the transportation regulations. In both the ANS/ANSI standard and the transportation regulations, the limit was selected because it is not possible to have a criticality accident involving fissile material at these low limits. Because it is not possible to have a criticality accident at the limits listed above, criticality accident alarm systems are not necessary for areas that comply with those limits.

The following Special Authorization has been identified in this license application:

- Surface Contamination Release Levels for Unrestricted Use – Items may be released for unrestricted use if the surface contamination is less than the levels listed in Table 4.6-1.

1.2.6 Security of Classified Information

USEC is required by 10 CFR 70.22(m) to submit, as part of its application for a license for the ACP, a plan describing the plant's proposed security procedures and controls, as set forth in 10 CFR Part 95, for the protection of classified matter. USEC satisfies the 10 CFR 70.22(m) requirements by submittal of the Security Plan for the Protection of Classified Matter as Chapter 2 of the Security Program for the American Centrifuge Plant. The Security Program is being submitted for NRC review along with this license application. In accordance with 10 CFR Part 95.15(b), USEC will submit, at least 60 days prior to operation of the ACP, an application for the transfer of Facility Clearance from DOE to the NRC.

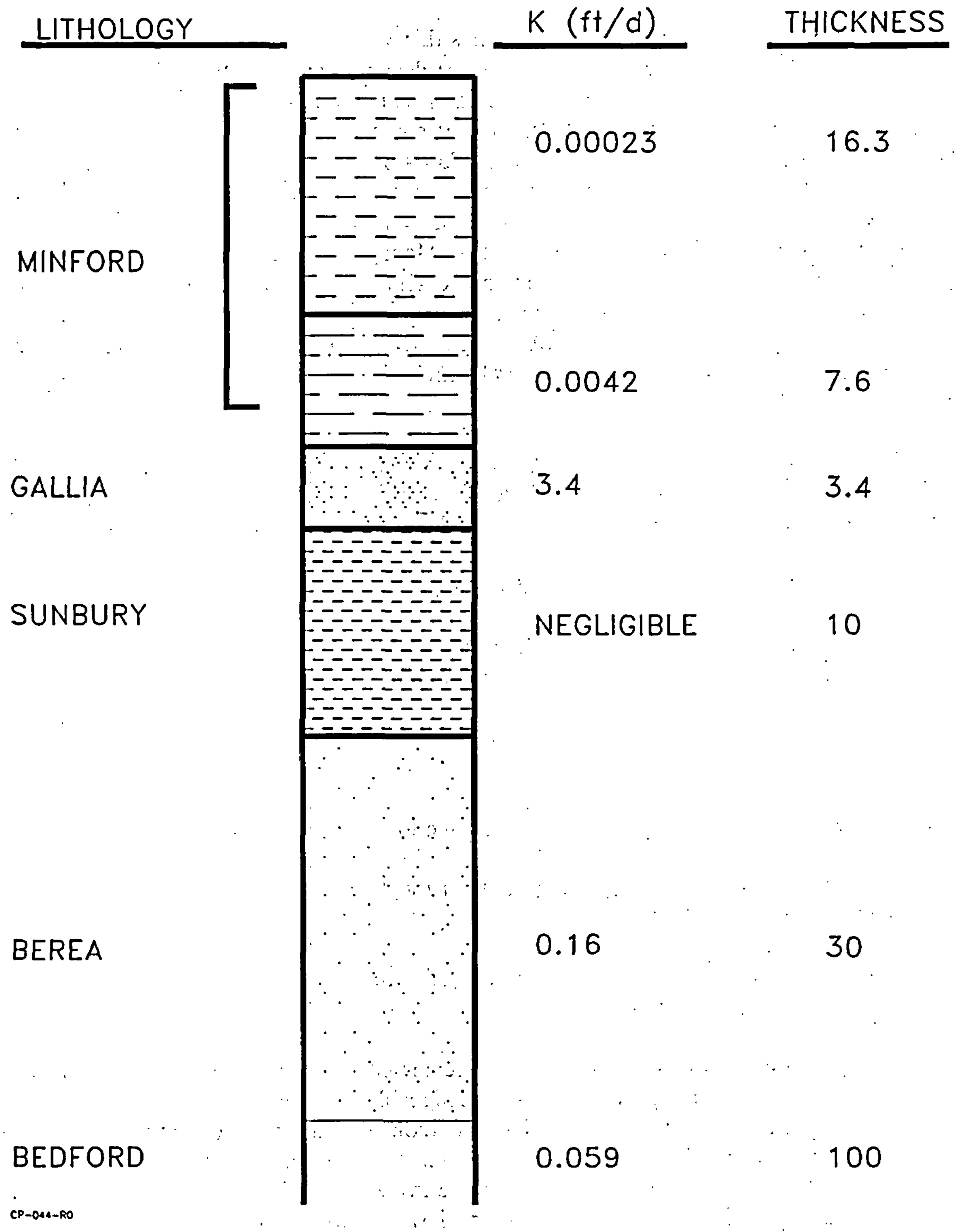

Figure 1.3-13 Geologic Column at the U.S. Department of Energy Reservation

1.4 Application Codes, Standards, and Regulatory Guidance

The ACP utilizes a number of the facilities that were originally constructed to support the GCEP and the GDP. The buildings/facilities were designed and constructed according to DOE requirements and/or nationally accepted codes and standards applicable at the time. Many of those codes and standards were earlier versions of current codes and standards that are utilized today for new construction. The codes and standards of record will be verified and documented during the ACP design verification process discussed in Section 11.1.6 of this license application. Any deviations from the codes and standards of record will be evaluated and documented in accordance with the Configuration Management Program as described in Section 11.1 of this license application. New buildings/facilities will meet the codes and standards applicable at the time the facility is designed and constructed as stated in plant design criteria. Modifications to existing buildings and/or facilities will be evaluated to determine if there is a safety benefit from applying current codes and standards and justification will be documented if current codes and standards are not applied.

The following sub-sections list the various industry codes, standards, and regulatory guidance documents that have been referenced in this license application. The extent to which USEC satisfies each code, standard, and guidance document is identified as follows:

1.4.1 American National Standards Institute/American National Society

- ANSI/ANS 3.2-1994, *Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants*

  USEC utilizes the provisions contained in Appendix A.6, paragraph (a) of this standard.

  For the reference to this standard, see Section 11.4.2.1 of this license application.

- ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactor*

  USEC satisfies the guidance of this standard with the following exceptions/clarification:

  > Section 4.1.6 - Operations are reviewed annually; however, personnel in the operating group who are knowledgeable of the NCS requirements for their operations perform this review. Personnel who are knowledgeable in NCS and are independent of operations (e.g., Engineering) provide assistance in these annual reviews. Personnel who are knowledgeable in NCS and are independent of operations (e.g., Engineering) review operations annually.

  For references to this standard, see Sections 5.4.1, 5.4.2, 5.4.5.1, and 5.4.5.2 of this license application.

- ANSI/ANS-8.3-1997, *Criticality Accident Alarm System*

  USEC satisfies the provision of this standard with the following exceptions/clarifications:

  > Section 1.2.5 – The primary radiation alarm system is the Criticality Accident Alarm System designed to detect a nuclear criticality and provide audible and visual alarms that will alert personnel to evacuate the immediate area. ACP primary facilities that handle ^{235}U in quantities greater than 700g have Criticality Accident Alarm System coverage except the UF_6 cylinder storage yards.

  For reference to this standard, see Section 5.4.4 of this license application.

- ANSI/ANS-8.19-1996, *Administrative Practices for Nuclear Criticality Safety*

  USEC satisfies the provisions of this standard with the following exceptions/clarification:

  > Section 7.8 - Operations are reviewed annually; however, personnel in the operating group who are knowledgeable of the NCS requirements for their operations perform this review. Personnel who are knowledgeable in NCS and are independent of operations (e.g., Engineering) provide assistance in these annual reviews. Personnel who are knowledgeable in NCS and are independent of operations (e.g., Engineering) review operations biannually (every two years).

  For references to this standard, see Sections 5.4.1 and 11.3.1.9 of this license application.

- ANSI/ANS-8.20-1991, *American National Standard for Nuclear Criticality Safety Training*

  USEC satisfies the provisions of this standard.

  For references to this standard, see Sections 11.3.1.1.2, 11.3.1.4, and 11.3.1.9 of this license application.

- ANSI/ANS-8.21-1995, *American National Standard for Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*

  USEC satisfies the provisions of this standard.

  For references to this standard, see Section 5.4.1 of this license application.

- ANSI/ANS-8.23-1997, *Nuclear Criticality Accident Emergency Planning and Response*

  USEC satisfies the provisions of this standard.

  For references to this standard, see Section 5.4.4 of this license application and Section 2.2.4 of the Emergency Plan for the American Centrifuge Plant.

E. Design outputs that consist of computer programs are developed, validated, and managed in accordance with ASME NQA-1-1994 Part II, Subpart 2.7, Basic Requirement 11.

F. Methods of design verification satisfy the provisions of Supplement 3S-1 of ASME NQA-1-1994.

G. Computer Program Testing is performed in accordance with ASME NQA-1-1994, Basic Requirement 11, "Test Control," and Supplement 11S-2, "Supplementary Requirements for Computer Program Testing."

H. Lifetime records are defined in accordance with ASME NQA-1-1994, Supplement 17S-1, "Supplementary Requirements for Quality Assurance Records," Section 2.7.1.

I. Hard copy or microfilm storage facilities satisfies the guidance of ASME NQA-1-1994, Supplement 17S-1, "Supplementary Requirements for Quality Assurance Records," Section 4.4.

For the references to this standard, see Section 11.5.1 of this license application and Sections 2.0, 3.0, and 11.0 of the QAPD for the ACP.

1.4.4 American Society of Mechanical Engineers

- ASME N509-1989, *Nuclear Power Plant Air-Cleaning Units and Components*

  New and existing fixed HEPA filter systems needed to ensure compliance with release limits or to control worker radiation exposure satisfy the provisions of this standard with the following exceptions/clarifications:

  Section 5.2 - Do not satisfy; No credit is taken for absorbers

  Section 5.5 - Do not satisfy requirements for air heaters

  Section 8.0 - Quality assurance requirements for applicable systems are identified in the QAPD

  Appendix A - Do not sample adsorbents

  Appendix B - Do not use allowable leakage guidance

  Appendix C – This appendix is used as guidance only

- NUREG/BR-0096, *Instruction and Guidance for Completing Physical Inventory Summary Reports, NRC Form 327*

  This NUREG provides line-by-line instructions for preparing NRC Form 327, Special Nuclear Material and Source Material Physical Inventory Summary Reports.

  USEC satisfies the provisions of this NUREG.

  For the reference to this NUREG, see Section 12.4 of the FNMCP for the ACP.

- NUREG/CR-4604, *Statistical Methods for Nuclear Material Management*

  This NUREG contains techniques and formulas used to estimate random and systematic error variances associated with nuclear material measurement methods.

  For the reference to this NUREG, see Section 9.1.1 of the FNMCP for the ACP.

- NUREG/CR-5734, *Standard Format and Content for the Fundamental Nuclear Material Control Plan Required for Low Enriched Uranium Enrichment Facilities*

  This NUREG is used to establish the Detection Quantity for evaluation of nuclear material inventory differences.

  For the reference to this NUREG, see Section 9.4 of the FNMCP for the ACP.

- NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*

  Portions of this NUREG were used as a general reference and guidance document in the development of the accident analyses in the ISA.

  For the reference to this NUREG, see Section 3.3 of the ISA Summary for the ACP.

- NRC Information Notice No. 88-100: *Memorandum of Understanding between NRC and OSHA Relating to NRC-Licensed Facilities (53 FR 43950, October 31, 1988)*, December 23, 1988

  USEC has reviewed the information contained in this Information Notice.

  For the reference to this IN, see Section 6.4 of this license application.

1.4.7 Other Codes, Standards, and Guidance

- Federal Guidance Report No. 11, "*Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*

The data contained in Tables 2-1 and 2-2 of this document used to calculate dose conversion factors for radionuclides of concern. This data is also used to calculate the Derived Air Concentrations (DACs) listed in Table 4.7-4.

For the reference to this guidance document, see Section 4.7.4 of this license application.

- American Society for Nondestructive Testing Recommended Practice No. SNT-TC-1A, June 1980 Edition

  USEC satisfies the provisions of this recommended practice.

  For the reference to this recommended practice, see Section 2.0 of the QAPD for the ACP.

- IAEA Safeguards Technical Manual, Part F, Volume 3

  The method used to establish sample sizes for item monitoring activities was obtained from this manual.

  For the reference to this recommended practice, see Section 7.4 of the FNMCP for the ACP.

- ANSI/IEEE 336-1985, *American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) Standard Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities*

  Periodic inspections and testing of items relied on for safety will be in accordance with Clause 7.

  For the reference to this standard see Section 2.6.8 of the ISA summary for the ACP.

- IEEE 1023-1988, *Institute of Electrical and Electronics Engineers (IEEE) Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations*

  The ACP design and operations are reviewed for human factors concerns. The ACP Human Factors Engineering program is performed in accordance with the requirements identified in IEEE 1023 with exception to Sections 6.1.1.12 and 6.1.1.18, which address mockup and simulation of new designs respectively. Also, exception is taken to any of the requirements in IEEE 1023 specific to nuclear power facilities.

  For the reference to this standard see Section 2.6 of the ISA summary for the ACP.

- ISA 67.04.01-2000 *Setpoints for Nuclear Safety-Related Instrumentation*

  The ACP IROFS related setpoints are determined utilizing methodologies in accordance with this standard.

  For the reference to this standard see Section 2.6.10 of the ISA Summary for the ACP.

1.5 References

1. NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*

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The minimum requirements for a qualified Senior NCS Engineer are:

- Completion of the minimum requirements for a qualified NCS Engineer;
- Performance of the functions of a qualified NCS Engineer;
- Completion of one year as a qualified NCS Engineer; and
- Approval by the NCS Manager (or equivalent).

The NCS Manager (or equivalent) may modify the minimum Senior NCS Engineer qualification requirements for personnel who have worked for a minimum of five years at other facilities as a nuclear criticality safety engineer.

5.3 Management Measures

5.3.1 Procedure Requirements

Operations to which NCS pertains are governed by written procedures or work packages. These procedures or work packages contain the appropriate NCS controls for processing, storing, and handling fissile material. The NCSE requirements that specify employee actions are incorporated into procedures or work packages as work instructions and are identified. Identifying these requirements ensures changes to these requirements are not made without review and approval by NCS. The NCSE requirements are incorporated into the appropriate procedures or work packages as required by the NCS Program procedure.

New and modified procedures or work packages are reviewed by the appropriate safety organizations, including NCS, as specified in the procedure for procedure control and/or work control process. NCS reviews the procedures and/or work instructions to verify that the appropriate NCSE requirements have been incorporated and to verify that the proposed operation complies with NCS Program requirements. Section 11.4 of this license application provides more details related to the procedure development and change process.

5.3.2 Posting and Labeling Requirements

Administrative NCS limits and controls for areas, equipment, and containers are presented through the use of postings and labels as specified in approved NCSEs and procedures. Postings and labels are proposed, reviewed, and approved during the NCSE review and approval process. Postings and/or labels are not required for engineered controls and may not be required for administrative controls when those limits and controls are included in "in-hand" operating procedures. These limits and controls are posted on

the NCS requirements signs as required by the plant NCS procedures. Approved NCSEs specify the wording for the postings. Labels are prepared in accordance with the plant NCS procedures and used as required by NCSEs. Limits and controls are printed or written in an appropriate size, and the postings and labels are placed in conspicuous locations such that they are legible to the operator at the work location, on the specific component, item, or piece of equipment, or posted at the entrance to an operating area or storage area. The specific locations may be specified in the applicable NCSE or determined by the supervision responsible for the material.

5.3.3 Change Control

A configuration management (CM) program ensures that any change from an approved baseline configuration is managed so as to preclude inadvertent degradation of safety or safeguards. The CM Program, described in Section 11.1 of this license application, includes organization and administrative processes to ensure accurate, current design documentation that matches the plant's physical configuration. The CM program applies to NCS and a change control process is utilized that helps ensure that the requirements of 10 CFR 70.72 are met, including the ISA Summary update requirements contained in 10 CFR 70.72(d)(3).

Functional and physical characteristics of operations controlled for NCS are described in NCSEs. Components and features that are identified in the NCSEs are analyzed to determine the "boundary" of the system, encompassing those interconnecting and/or supporting items that are essential to ensure availability and reliability. The boundaries are identified on system drawings, and the configuration is verified to be as-built. These components and features are maintained in a design control document for the building or process. Each time a change is planned, the document is reviewed by the individual (e.g., design authority, systems engineer, operations manager, maintenance, etc.) planning the change to determine if the change affects an IROFS. The NCS Program establishes and maintains NCS safety limits and NCS operating limits for IROFS in nuclear processes and maintains adequate management measures to ensure the availability and reliability of the IROFS.

The change control process specifies the organizations required to perform reviews of changes. If an item is relied on for the criticality safety of an operation (i.e., is an IROFS), it will be identified and NCS reviews the NCSE for the specific operation and determines if the change affects the analysis performed and the conclusions made in the NCSE. The change request will be approved by NCS only if the change does not adversely impact NCS, or once a revised NCSE has determined that the change is acceptable and meets NCS Program requirements. If a change affects the ISA Summary, it is updated appropriately. In this way, modifications to controlled operations are evaluated and approved prior to implementation and placing the affected structures, systems, or components in service.

Records management and document control (RMDC) is another element of CM and is described in Section 11.7 of this license application. Procedures, documents, and records control programs provide for centralized control and issuance of documents essential to the maintenance of the design history, and a repository for records to verify this maintenance. NCSEs are specifically included in the index of documents that are required to be controlled.

5.3.4 Operation Surveillance and Assessment

To ensure that the NCS Program is properly established and implemented, walk-throughs, assessments, and audits are utilized.

Operating SNM process areas are reviewed on a regular basis through a combination of walk-throughs and reviews by work crew supervision. NCS walk-throughs of facilities that may contain fissile material operations are performed by NCS personnel to determine the adequacy of implementation of NCS requirements and to verify that conditions have not been altered to adversely affect NCS. These walk-throughs are performed as specified by the NCS procedure on walk-throughs. For example, a walk-through inspection can be performed in response to trend data, at the request of the operations personnel, or due to concerns raised by employees or NCS personnel. As a minimum, specific fissile material operating areas are assessed by NCS personnel via walk-through at least annually, sometimes in conjunction with the assessments discussed below. By distributing the various areas' walk-throughs over a year's time, NCS personnel are performing a field walk-through on approximately a monthly basis.

Work crew supervision provides real-time assessments of fissile material operations within their operating area to ensure NCS requirements are being adequately implemented and operating conditions have not been altered to adversely affect NCS. Fissile material operations management also performs an annual self-assessment to ensure NCS program requirements are being met in the field.

In addition to the annual self-assessments, independent internal audits of the NCS Program are conducted or coordinated by the Quality Assurance Manager as described in Section 11.5 of this license application. The purpose of these audits is to determine the adequacy of the overall NCS Program. This includes the adequacy of the NCSEs, internal assessment programs, and implementation of the NCS requirements.

The results of these walk-throughs, assessments, and audits are documented and reported to appropriate management.

If a condition is identified that is non-compliant with NCS program requirements, field personnel are to report the condition as directed by plant procedures. If the condition is not covered by an existing procedure, consultation with a qualified NCS engineer is required before taking any corrective action. Immediate corrective actions may be provided by the responding NCS engineer verbally or in writing. NCS emergency response is discussed in Section 5.4.2 below.

Managers in charge of fissile material operations are provided additional training on NCS and response to NCS deficiencies as described in Section 11.3.1.4 of this license application. NCS deficiencies are reported in accordance with the requirements contained in 10 CFR Part 70, Appendix A or other appropriate reporting requirements. Incident reporting and investigation is described in Section 11.6 of this license application. The deficiency data is trended to monitor and prevent future violations. Corrective actions are taken for adverse trends in accordance with the Quality Assurance Program Description for the American Centrifuge Plant and the Corrective Action Program as described in Section 11.6.7 of this license application, and records of actions taken are retained in accordance with RMDC requirements described in Section 11.7 of this license application.

5.4 Methodologies and Technical Practices

5.4.1 Adherence to American National Standards Institute/American Nuclear Society Standards

The NCS Program has been developed to comply with the American National Standards Institute (ANSI)/American Nuclear Society (ANS) ANSI/ANS-8.1-1998, ANSI/ANS-8.19-1996, and ANSI/ANS-8.21-1995 standards as discussed in this section.

5.4.2 Process Evaluation and Approval

Each operation involving uranium enriched to 1 wt. percent or higher ^{235}U and 100 g or more of ^{235}U is evaluated for NCS prior to initiation. The evaluation describes the scope of the operation, evaluates credible criticality accident contingencies, and establishes NCS requirements to maintain the operation subcritical. The evaluation process is governed by written procedures.

When an NCSE (or a change to an existing NCSE) is needed for a particular fissile material operation, a request is submitted to the NCS group to evaluate the proposed operation. Other methods for initiating an NCS change include, but are not limited to: 1) the engineering change process, and 2) the corrective actions process, self-assessments, and external audits and inspections.

In response to the request, an NCS evaluation may be performed or the request may be returned due to inadequate detail, the change is bounded by a current analysis, or the operation does not involve uranium enriched to 1 wt. percent or higher ^{235}U and with mass of 100 g or more ^{235}U (see Section 5.4.2.1). If necessary, a NCSE is prepared (or an existing NCSE is revised) to document the analyses performed as specified in the NCS evaluation procedure. A hazard identification process (e.g., a "What-If" analysis) is used to identify and document potential upset conditions, or contingencies, presenting NCS concerns. Engineering judgment of the qualified NCS engineer may indicate the need for a more detailed study. For example, a hazards and operability study may be used if the operation is complex and involves multiple interacting systems that require substantial input from operations, maintenance, and other subject matter experts to identify the possible upset conditions. A contingency analysis is performed in which the subcriticality of a process, given the occurrence of the contingency, is assessed. This analysis demonstrates the double contingency principle for the proposed operation.

The double contingency principle as stated in ANSI/ANS-8.1-1998, Section 4.2.2, is: "Process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." The ACP NCS Program meets the double contingency principle by implementing at least one control on each of two different parameters or implementing at least two controls on one parameter. Controls include passive engineered barriers (e.g., structures, vessels, piping, etc.); active engineered features (e.g., valves, thermocouples, flow meters, etc.); reliance on the natural or credible course of events (e.g., relying on the nature of a process to keep the density of uranyl fluoride less than a specified fraction of theoretical); and administrative controls that require performance of human actions in accordance with approved procedures or work instructions, or by other means that limit parameters

within specified values. If two controls are implemented for one parameter, the violations or failure scenarios addressed by the controls will be independent. Application of this principle ensures that no single credible event can result in an accidental criticality or that the occurrence of events necessary to result in a criticality is not credible.

The NCSE will document the basis for the conclusion that a change in a process or parameter is "unlikely". The basis may be an engineered feature, administrative control, the natural or credible course of events, or any combination of these or other means necessary to ensure the change is unlikely to occur. The parameters or conditions relied on and the limits must be specified in the NCSE and controlled.

Where the natural or credible course of events is relied upon in whole or in part to prevent a process condition change, the factors that influence the process are described in sufficient detail in the NCSE as items related to NCS and programmatically controlled. For items that are established, maintained, and implemented by non-NCS programs, credit for availability and reliability is established as described in Section 11.1 of this license application without the need for additional NCS controls. For situations where the NCS-credited controls do not provide adequate assurance of availability or reliability (i.e., situations where non-NCS programmatic and physical plant changes could adversely affect the intended criticality safety function of the items relied upon for criticality safety), specific NCS controls are established, maintained, and implemented to ensure criticality safety.

The NCS evaluation process involves a review of the proposed operation and procedures or work instructions, discussions with the subject matter experts to determine the credible process upsets which need to be considered, development of the controls necessary to meet the double contingency principle, and identification of the assumptions and equipment (i.e., physical controls) needed to ensure criticality safety.

Engineering judgment of both the analyst and the technical reviewer is used to ascertain independence of events and their likelihood or credibility. The basis for this judgment is documented in the NCSEs. Depending on the complexity of the operation, analytical methods such as Fault Tree and Event Tree Analyses may be used in the evaluation process to examine potential accident scenarios. When needed to support the analytical method, qualitative or quantitative estimates of event frequency are developed to support the determination of the likelihood of an event.

Once the NCSE is completed, a technical review of the evaluation is performed and documented. The technical review of an NCS evaluation is performed by a Senior NCS Engineer or is a NCS Engineer completing the technical review under the guidance of a Senior NCS Engineer.

The NCSE documents the NCS requirements for the operation. The NCS requirements include the process conditions that must be maintained to meet the double contingency principle or preserve the documented basis for criticality safety and restrict the modes of operation to those that have been analyzed in the NCSE. The requirements to be included in operating procedures and/or work instructions, and postings are identified.

The NCSE approval process first involves the acceptance of the NCSE by the technical reviewer. A review is then performed by the NCS Manager to ensure consistency with other NCSEs and other potentially conflicting requirements or regulations. After approval by the NCS Manager, a review is performed in accordance with 10 CFR 70.72 as described in Section 11.1.4 of this license application to determine whether prior NRC approval of the NCSE is required. If NRC approval is not required, the NCSE is reviewed by the responsible organization manager. Editorial changes require only the approval of the NCS Manager. Editorial changes are defined as changes that do not change the technical basis of the NCSE. Once approved, the NCS controls, limits, evaluation assumptions, and safety items are verified to be fully implemented in the field. The operations organization and NCS personnel perform this verification process. The documentation of this verification process is maintained as a quality record along with the NCSE.

Management of the operating organization is responsible for implementing, through training and procedures or work instructions, the conditions delineated in the NCSE. Operational aids such as postings, labels, boundaries for fissile material operations, and fissile material movement guidelines are provided as specified in the NCSE. The manager/supervisor ensures postings and labels are prepared and verify that they are properly installed as required by the NCSE. The procedures and/or work instructions are prepared or modified to incorporate the NCSE requirements. Managers/supervisors are responsible for ensuring the employees understand the procedures and/or work instructions and understand the NCS requirements before the work begins.

Each completed NCSE is issued as a controlled document. Completed NCSEs are archived and retrievable as permanent quality records in accordance with the RMDC requirements described in Section 11.7 of this license application. The NCSE process provides assurance that operations will remain subcritical under both normal and credible abnormal conditions.

Emergencies arising from unforeseen circumstances can present the need for immediate action. If NCS expertise or guidance is needed immediately to avert the potential for a criticality accident, direction will be provided orally or in writing. Such direction can include a stop work order or other appropriate instructions. Documentation will be prepared within 48 hours after the emergency condition has been stabilized.

New operations must comply with the double contingency principle.

5.4.2.1 Non-Fissile Material Operations

Some operations involve situations in which the uranium has an enrichment of less than 1 wt. percent ^{235}U or an inventory of less than 100 g ^{235}U. These operations are termed "non-fissile material operations" and are performed without the need for NCS double contingency controls. The determination of which operations are fissile versus which operations are non-fissile may be contained within a NCSE or as a separate document. When the determination is outside a NCSE, the determination need not be performed by a qualified NCS Engineer. The determination of an operation being non-fissile must include normal and credible abnormal upset conditions to ensure the enrichment and/or inventory are maintained below 1 wt.

percent ^{235}U or below 100 g ^{235}U. Controls are sometimes applied to a non-fissile material operation to ensure it does not inadvertently involve fissile material. These controls can be either engineered or administrative and may be incorporated into applicable operating procedures or work instructions when it is determined they are needed to maintain the non-fissile material operation below either 100 g ^{235}U or 1 wt. percent ^{235}U. This determination is made by the responsible line manager.

5.4.3 Design Philosophy and Review

Through the CM Program, designs of new fissile material equipment and processes must be approved by NCS before implementation. Where practical, the use of engineered controls on mass, geometry, moderation, volume, concentration, interaction, or neutron absorption will be used as the preferred approach over the use of administrative controls. Advantage will be taken of the nuclear and physical characteristics of process equipment and materials, provided control is exercised to maintain them if they may credibly degrade such that control of the parameter is jeopardized.

The preferred design approach includes two goals. The first is to design equipment such that NCS is independent of the amount of internal moderation or fissile concentrations, the degree of interspersed moderation between units, or the thickness of reflectors. The second is to minimize the possibility of accumulating fissile material in inaccessible locations and, where practical, to use favorable geometry for those inaccessible locations. The adherence to this approach is determined during the preparation and technical review of the NCSE performed to support the equipment design. This preferred design approach is implemented as described in NCS procedures.

Fissile material equipment designs and modifications are reviewed to ensure that engineered controls are used for NCS to the extent practical. Administrative limits and controls will be implemented to satisfy the double contingency principle for those cases where the preferred design approach is not practical.

5.4.4 Criticality Accident Alarm System Coverage

A criticality accident alarm system (CAAS) that complies with 10 CFR 70.24 and ANS/ANSI-8.3-1997 is provided to alert personnel if a criticality accident occurs. The system utilizes an audible and/or visual signal to alert personnel in the area to evacuate to reduce radiation exposure resulting from the incident.

The need for CAAS coverage is considered during the development process for NCS evaluations. In general, coverage is provided for fissile material operations, except the UF_6 cylinder storage yards as specified in Section 1.2.5 of this license application. Other exceptions to CAAS coverage are documented in NCS evaluations and are based on a conclusion in the NCSE that a criticality accident is non-credible in the area where the fissile material operation is ongoing. Conclusions of non-credibility require at a minimum that the inventory of ^{235}U in the area is less than 700 g, less than 50 g per square meter, or less than 5 g in any 10 liter volume and that it is non-credible for these values to be exceeded (see Section 1.2.5 for a description of the exemption to the requirements of 10 CFR 70.24). In addition, CAAS is not required for areas having material that is either packaged or stored in accordance with 10 CFR Part 71 or specifically exempt according to 10 CFR 71.53. Areas that do not contain fissile material operations do not require a NCSE and do not require CAAS coverage.

The CAAS is designed to detect neutron radiation levels that would result from the minimum criticality accident of concern as defined by ANSI/ANS 8.3-1997 and to provide an audible evacuation alarm. A secondary function is to activate the building radiation warning lights and alarms at the X-3012 Process Support Building Area Control Room (ACR) and the X-1020 Emergency Operations Center.

For each area requiring CAAS coverage, a monitoring system is installed that provides coverage of the area by at least two independent detection units, each with the ability to actuate the alarm. This arrangement allows for one detection unit to be temporarily out of service with fissile operations continuing under the coverage of the other detection unit. A detection unit is a set of at least three neutron sensitive radiation detectors that may be co-located or may be distributed over the area. The detection logic of the system requires that two of the three neutron detectors must be activated to initiate the building evacuation alarm system. Each detector may be logically part of more than one detection unit.

The building evacuation alarm system includes interior evacuation horns and exterior radiation warning lights to deter personnel from re-entering the building after an evacuation. In addition, facilities within 200 feet of a building/facility requiring CAAS coverage have radiation evacuation horns installed inside and radiation warning lights installed on the exterior. Personnel who have routine access to these facilities have been trained to recognize and respond to these indications as described in Section 11.3.1.1.2 of this license application.

To protect against the loss of coverage, the CAAS includes redundant decision logic, a backup power supply, detector status information and system self-diagnostic information are provided to the X-3012 building ACR and X-1020 building. The CAAS has been designed to survive and/or withstand credible abnormal events as described in the accident analysis for a sufficient time to warn personnel to evacuate. In the event CAAS coverage is lost for an operation, plant procedures provide for compensatory actions, which may include shutdown of equipment, limiting access, halting movement of uranium-bearing material, or other actions.

Additional information provided by the CAAS includes a historical log of events and the capability to monitor and record the criticality accident for managing the post-accident situation and any remedial action. Nuclear accident planning and response is discussed in Section 2.2.4 of the Emergency Plan for the American Centrifuge Plant.

5.4.4.1 Portable CAAS

In the event a fissile material operation requiring CAAS coverage is performed beyond the detection range of established CAAS instrumentation, a portable unit may be used. The portable unit has the same detection capabilities as the permanently installed units, although those capabilities may be based on gamma radiation. Alarm annunciation, however, is usually limited to the immediate area within the audible range of the unit's alarm with an additional telemetric link to the X-3012 ACR and X-1020. This link will transmit the location of the unit, if mobile, and allow the use of the plant PA system to warn personnel within 200 feet of the area of the portable unit to evacuate. A

portable unit may only be used on a temporary basis and it may be located indoors, outdoors, or on a vehicle.

5.4.5 Technical Practices

5.4.5.1 Application of Parameters

Moderation

Water is considered to be the most efficient moderator commonly found in the ACP. When moderation is not controlled either optimum moderation or worst credible moderation is assumed as the normal case when performing analyses. When moderation is controlled, credible abnormal process upset conditions determine the worst-case moderated conditions. Generally, moderation control is not maintained by measurement; however, when used, dual independent sampling methods are implemented.

Moderation control is applied to plant equipment containing UF_6. In areas where greater than the safe mass of uranium (as defined below) is handled, processed, or stored and moderation controls are applied, restrictions are placed on firefighting procedures to limit the use of moderator material. However, even in these areas, the application of the double contingency principle ensures the worst credible loss of moderation control cannot result in a critical configuration without an additional independent and concurrent upset event.

The centrifuge process equipment is comprised of a variety of closed systems designed to process gaseous UF_6. This closed system prevents the introduction of moderation due to wet air in-leakage. Also, because UF_6 reacts chemically with moisture (a moderator) to produce solid uranium-bearing compounds that impedes the proper operation of the process equipment, the UF_6 bearing systems are designed to minimize introduction of moisture.

Volume

Volume limits are used as specified in NCSEs. The bases for volume limits are provided in each NCSE prepared for those operations requiring containers. Specific details of these bases can be obtained by referring to the applicable NCSE. When volume control is used, the size of the containers is ensured through the CM Program and/or by procedurally requiring the use of certain containers for fissile material operations.

Interaction

Interaction is controlled by spacing items bearing fissile material when those items could result in a criticality accident if not properly spaced. The spacing necessary to maintain a safe array of fissile material units is determined in the NCSE performed for the array. The amount of spacing needed between items is determined based on analysis of the normal and credible abnormal process upset conditions for the particular operation. The basis for the spacing is documented in NCSEs. In accordance with the preferred design approach, described in Section 5.4.3 of this chapter, passive

engineered controls are used to the extent possible to ensure spacing requirements are maintained. When used, the structural integrity of the spacers or racks is sufficient to maintain spacing for normal and credible abnormal upset conditions.

Geometry

Geometry control is applied by limiting equipment dimensions for those systems that depend on the geometry for criticality safety. The geometry is determined in the NCSE that is performed for each system and depends on the normal and credible abnormal process upsets conditions related to the specific system. Geometry controls are specified in the NCSEs, are maintained by the CM Program, and are verified prior to authorizing initial operation. Safe geometry dimensions may be obtained from established standards or operation specific reactivity calculations.

Mass

Mass controls are applied on a case-by-case basis depending on the fissile material operation involved. The acceptable mass is determined based on the specific NCSE performed for the operation. The safe mass value depends on many factors including the geometry, the ^{235}U enrichment, composition, etc. Safe mass values may be obtained from established standards or operation specific reactivity calculations. Experimental data is not used as the sole source for safe mass values. Safe mass values are chosen to ensure no single credible upset can result in a critical configuration. When safe mass values are dependent on the geometry, enrichment, composition, or some other parameter, the combination of mass and the other parameter is used as one control to meet the double contingency principle. The safe mass values are communicated to the operating personnel via the operating procedures and/or work packages.

Unless specifically controlled, an item containing enriched uranium is assumed to contain the most ^{235}U credible based on the available volume. When mass is determined through measurement, instrumentation is used.

Enrichment

Uranium-containing material in the ACP with ^{235}U enrichment less than 1 wt. percent is considered incapable of supporting a nuclear chain reaction, but interaction of such materials with materials of higher enrichment is taken into consideration in the specific NCSE for those operations which involve material enriched to greater than 1 wt. percent.

The maximum ^{235}U enrichment of UF_6 in the ACP is 10 wt. percent. Small quantities of greater than 10 wt. percent ^{235}U may be present outside of plant equipment in the form of laboratory samples or standards. Some buildings on the reservation may be used to process and/or store fissile material from both the ACP and Portsmouth Gaseous Diffusion Plant (GDP). Although the GDP has historically processed material at greater than 10 wt. percent ^{235}U, this material is no longer readily available to interact with ACP operations. However, for conservatism, some operations in these common buildings may be analyzed at greater than 10 wt. percent ^{235}U enrichment.

The maximum ^{235}U enrichment for each operation is established by the specific NCSE. The NCSE specifies the maximum acceptable enrichment for each operation. Credible process upset conditions that could alter the ^{235}U enrichment are also considered in the NCSEs. Due to the difficulty in obtaining reliable, real-time enrichment measurements that are both accurate and precise enough to use as a NCS control, enrichment is assumed to be the maximum credible for each operation. When the enrichment of uranium needs to be measured for a NCS control, the measurement is obtained using either installed equipment or based on samples analyzed in a laboratory.

Density

The density of materials used in a given operation is justified in the NCSE for the operation being considered. If the density must be controlled to maintain compliance with the double contingency principle, it will be documented in the specific NCSE for the operation and it will be measured using instrumentation.

UF_6 in the gaseous phase, at any credible pressures and temperatures existing in the plant equipment, is incapable of supporting a nuclear chain reaction even when intermixed with hydrogenous material (e.g., hydrogen fluoride [HF]). UF_6 in the gaseous phase in plant equipment has low material density.

Heterogeneity

Heterogeneous configurations are considered for those operations that involve small fissile material and moderator regions. Heterogeneous groupings may occur for the handling of small sample containers; however, 10 wt. percent ^{235}U is assumed for samples handled on a safe mass basis. Using the homogeneous safe mass of 10 wt. percent ^{235}U is also safe for heterogeneous 10 wt. percent ^{235}U because, at this enrichment, the homogeneous and heterogeneous minimum critical masses are close in value.

Concentration

Concentration controls are used on a case-by-case basis. When the criticality safety of an operation depends on the concentration of fissile material, the medium is sampled twice, the samples are verified to be properly taken by a second individual, and the two samples are independently analyzed as required by the specific NCSE for the operation involved. The specific controls and details are documented in the NCSE for each operation that relies on concentration controls. No operations exist at the plant where concentration control is applied to an operation involving more than a safe mass of uranium. A container with concentration controlled solution is kept normally closed. Precipitating agents, including freezing, are controlled as necessary to ensure they do not inadvertently increase the concentration.

A typical operating limit is 5 g ^{235}U per liter, regardless of enrichment. A concentration of 11.6 g ^{235}U per liter is considered subcritical at any enrichment, as recognized by ANSI/ANS-8.1. If, under all postulated conditions, the concentration is always less than 11.6 g ^{235}U per liter, the operation is considered subcritical.

Reflection

Normal and credible abnormal reflection is considered when performing NCS evaluations. The possibility of full water reflection is considered when performing analyses. It is recognized that concrete can be a more efficient reflector than water, and its potential presence is considered. Reflection controls are used to limit the potential reactivity of a fissile material operation.

Neutron Absorption

When neutron absorbers are used as NCS controls, the intended distributions and concentrations under both normal and credible abnormal conditions are maintained in accordance with the requirements of the applicable NCSE and ANSI/ANS-8.21-1995. These requirements are: representative sampling of the neutron absorber, sampling at a frequency based on the environment to which the neutron absorber is exposed, analyzing of samples for all material attributes for which credit is taken in the NCSE, and periodic inspections of fixed neutron absorbers to ensure adequate distribution as specified in the NCSE.

A NCS evaluation can take credit for the neutron absorption properties of the materials (1) added specifically for the purpose of absorbing neutrons, and (2) of construction, provided an allowance has been made for manufacturing and dimensional tolerances, corrosion, chemical reactions, neutron spectra, and uncertainties in the neutron cross-sections.

5.4.5.2 Methods of Calculation

Experimental Data

Experimental data are not specific enough to allow evaluation of operations performed in the ACP. The generic nature of the experimental data does not address the variables present in the different operations. However, experimental data are used for validation of the computer code (e.g., KENO V.a) used to perform the calculations needed to support the development of NCSEs. The experimental data used are discussed in the code validation report (Reference 11).

Handbooks

Handbooks are also used in some cases when simple systems are being evaluated. Most of the operations performed in the ACP are too complicated to be adequately addressed by data in a handbook. When isolated operations are performed with small amounts of fissile material, referencing handbooks is useful to support conclusions in the NCSE. Examples of the handbooks used include, but are not limited to, ARH-600, *Criticality Handbook* and LA-10860-MS, *Critical Dimensions of Systems Containing* ^{235}U, ^{239}Pu, *and* ^{233}U.

Hand Calculations

Applicable methods for evaluating single units include Modified Two Group Diffusion Equation (i.e., Critical Equation), Buckling Conversion, and Comparative Analysis.

- **Modified Two Group Diffusion Equation** – This method is applicable to, and most widely used for, solution systems.
- **Buckling Conversion** – The method of buckling conversion or shape conversion is applicable to all materials.
- **Comparative Analysis** – This method involves direct comparison of the system configurations to subcritical data from NCS handbooks.

Applicable methods for evaluating arrays include the Solid Angle Method and the Surface Density Method using unit shape factor.

- **Solid Angle Method** – This method is applicable to solution systems. It is not useful if reflection is more effective than a thick water reflector located at the array boundary. The conditions that must be satisfied in order to successfully apply the solid angle method are (1) $k_{effective}$ (k_{eff}) of any unreflected unit does not exceed 0.80; (2) each unit is subcritical when completely reflected by water; (3) the minimum surface-to-surface separation between units is 0.3 meters; and (4) the allowed solid angle does not exceed 6 steradians.
- **Surface Density Method** using unit shape factor – This method can be used as an approximation for large arrays of identical units containing solutions and metals. This method determines the spacing and mass of units independent of the number of units. An important feature of the Surface Density Method is that it is equally applicable to more irregular geometries.

When hand calculations are used, the specific methodology employed will be as described in "Nuclear Criticality Safety" by R.A. Kneif, American Nuclear Society, 1991 and subject to a total system reactivity of 0.95 for all credible off normal events.

Computer Calculations

For those cases where adequate references are not available, NCS computational analyses are performed, which involve the calculation of k_{eff} to determine whether the system will be subcritical under both normal and credible abnormal process conditions. Computer codes that simulate the behavior of neutrons in a process system or that solve the Boltzmann transport equation are used.

Computer calculations of k_{eff} provide a method to relate analytical models of specific system configurations to experimental data derived from critical experiments. A critical experiment is defined as a system that is intentionally constructed to achieve a self-sustaining neutron chain reaction or criticality. Critical experiments that have specific, well-defined parametric values and are adequately

documented are termed benchmark experiments. Computer codes are validated using experimental data from benchmark experiments that, ideally, have geometries and material compositions similar to the systems being modeled.

Validation of the computer code determines its calculational bias or uncertainty as well as the effective margin of subcriticality. The validation involves the modeling of benchmark critical experiments over a range of applicability. Because the k_{eff} value of a critical experiment is essentially 1, the bias of the code is taken to be the deviation of the calculated values of k_{eff} from unity. Statistical analysis is employed to estimate the calculational bias, which includes the uncertainty in the bias and uncertainties due to extensions of the area of applicability, as well as the effective margin of subcriticality. Uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of the experimental data. The validation of the computer code specifically defines the maximum acceptable k_{eff} used to determine subcriticality.

The margin of subcriticality used for the plant results in a k_{eff} upper safety limit that ensures that there is a 95 percent confidence that 99.9 percent of future k_{eff} values less than this limit will be subcritical. The minimum margin of subcriticality of 0.02 in k_{eff} is used to establish the acceptance criteria (i.e., upper safety limit) for criticality calculations. The upper safety limit varies with the computer system, codes, cross sections, and materials used in the validation.

The calculation of k_{eff} is accomplished by the use of computer codes that utilize Monte Carlo techniques to determine k_{eff} of a system. Computer models representing the geometrical configuration and material compositions of the system are developed for use within the code. The development of appropriate models must account for or conservatively bound both normal and credible abnormal process conditions.

When NCS is based on computer code calculations of k_{eff}, controls and limits are established to ensure that the maximum k_{eff} complies with the applicable code validation for the type of system being evaluated. For example, NCS related IROFS developed during initial license application were developed using reactivity calculations performed on personal computers running the Microsoft Windows XP operating system and validated as described in Reference 11 with an upper safety limit of 0.955. Reactivity calculations, performed after initial license application, comply with the code validation for the specific system used to perform the calculation.

Scoping and analysis calculations may be performed utilizing various unvalidated computer codes; however, computer calculations of k_{eff} used as the basis for NCS evaluations are confirmed by, or performed using, configuration-controlled codes and cross-section libraries for which documented validations are performed with at least the same degree of conservatism as that presented in the validation report WSMS-CRT-03-0093, Revision 0, November 2003, and are in accordance with ANSI/ANS-8.1-1998.

The computer codes and cross sections used in performing k_{eff} calculations are maintained in accordance with a configuration control plan. Quarterly, or prior to use, one of the following is performed: a bit-by-bit comparison of the production version of the software (executable modules and data libraries) versus an archived production version; or a comparison of the output from all validation cases versus archived output of all validation cases from the original validation performed when the production version was installed to ensure no changes in the calculated k_{eff} for the validation cases. Changes to the hardware or software are evaluated in accordance with 10 CFR 70.72 change requirements. The System Administrator, a NCS engineer, is responsible for controlling access to the software.

10.2 Decommissioning Steps

Decommissioning may begin immediately following termination of operation, since only low radiation levels exist at this plant. Overall, the decommissioning is estimated to require approximately six years from plant shutdown to completion of the final status survey of radiological conditions. The order of activities to support decommissioning will generally be: planning and preparation; process system purging; equipment dismantling and removal; decontamination; disposition of equipment and material (including classified items); disposal of wastes; completion of a final status survey. The following sections provide an overview and explanation of each of these steps.

10.2.1 Overview

The intent of decommissioning is to return the ACP to an unrestricted use state. Removed equipment includes the centrifuges, the feed and withdrawal equipment, piping and components from systems providing UF_6 containment, systems in direct support of the centrifuges (e.g., cooling water), radioactive and hazardous waste handling systems, contaminated air filtration systems, etc. The remaining plant infrastructure includes utility services such as electrical power supply, sanitary water, fire suppression, ventilation, communications, and sewage treatment.

Decontamination of the plant will not require the installation of a new facility dedicated for that purpose since the X-7725 facility will serve as the DSA and will accommodate repetitive equipment decontamination of centrifuges and other components. The DSA is described in Section 10.8.1 of this license application and will be the location for decontamination activities.

Although certain unclassified components may be reused or sold as scrap, for conservatism this plan assumes only that components will be decontaminated in accordance with radiation protection requirements. Classified parts will be dispositioned in accordance with the Security Program. Table 10.2-1 of this license application lists components for potential decontamination at decommissioning.

USEC intends to evaluate possible commercial uses of UF_6 tails. UF_6 tails which are not commercially reused will be converted to a stable form and disposed of in accordance with the USEC Privatization Act and other applicable statutory authorizations and requirements at DOE's UF_6 conversion facilities and/or other licensed facilities. UF_6 tails are stored in steel cylinders until the tails material can be processed in accordance with the disposal strategy established by USEC. USEC provides financial assurance to fund the estimated cost of conversion and disposal of the depleted uranium inventory as it is generated during operation. This funding is described in the DFP and is in addition to the funding requirements for decommissioning the ACP. At full capacity, the ACP will generate approximately 9,520 Metric Ton (MT) of UF_6 tails annually. Over the 30-year license, that is a total of approximately 265,300 MT of UF_6 tails, as noted in Table C3.19 of the DFP. Depending on technological developments and the existence of facilities available prior to ACP shutdown, the tails may have commercial value and may be

marketable for further enrichment or other processes. However, funding provisions are made to dispose of the tails should that become necessary.

Contaminated portions of the buildings will be decontaminated. Structural contamination is expected to be limited to the areas indicated on Figure 10.1-1 (located in Appendix A) inside the CCZ of the plant. The remainder of the ACP is not expected to require decontamination. Good housekeeping practices during normal operation and cleanup activities following spills or contamination events will maintain these other areas contamination free. Decontamination activities will continue until facilities satisfy the specified radiological criteria.

10.2.2 Purging

At the end of useful operation, the ACP is shut down and UF_6 material is removed to the fullest extent possible by normal process operation. This is followed by evacuation and purging of process systems. This shutdown and purging portion of the decommissioning process is estimated to take approximately three months.

10.2.3 Dismantling and Removal

Dismantling is the process of unbolting, disconnecting, cutting, etc., of components requiring removal. The dismantling and removal activities are simple but labor intensive. They generally require the use of protective equipment. The work process will be optimized, considering the following:

- Minimize spread of contamination and the need for protective equipment;
- Balance the number of cutting and removal operations with the resultant decontamination and disposal requirements;
- Optimize the rate of dismantling with the rate of decontamination plant throughput;
- Provide storage and laydown space required, as impacted by retrievability, criticality safety, security, etc.; and
- Balance the cost of decontamination with the cost of disposal.

Details of the complex optimization process will be decided near the end of plant useful life, taking into account specific contamination levels, market conditions, and available waste disposal sites. To avoid laydown space and contamination problems, dismantling will proceed generally no faster than the downstream decontamination process. The time frame to accomplish both dismantling and decontamination is estimated to be five years.

Table 10.2-1 Components for Potential Decontamination at Decommissioning

Components	Description [units]	Estimated Quantity
Centrifuges	Internals: Rotor Assemblies, Motors, Suspensions and Mounts (Classified)	12,000[1]
Piping	1 to 10 inch process piping length (Lft)	168,100
Pumps	Vacuum Pumps (Evacuation/Purge)	246
Ventilation	Ductwork; Miscellaneous Gulper Ducting (ft^3);	118
Surface Areas[2]	Building Floors, Yards, Equipment (ft^2)	2,795,642
Valves	Process valves (excluding Sheetmetal)	7,250
	Miscellaneous valves	652
Process Equipment	**[This information has been withheld pursuant to 10 CFR 2.390]**	
Scales	Process Weighing Equipment	6
Compressors	Process Gas Compressors	12
Heat Exchangers	Machine Cooling Water HX, Freezer/Sublimers Compressor Train Coolers	16
Traps	Chemical traps (8 banks of 4), Cold Traps, Roughing Filters, Miscellaneous Traps	111
Tanks	Mixing, Holdup, Surge, and Dump Tanks	15
Cylinders	Tails (14, 10 Ton)	21,269
Cylinders	Tails, Parent (2.5 Ton)	1,000
Other Equipment	UF_6 Portable Carts, Buffer Storage Stands, and Gas Test Stand Equipment (Valve boxes)	66
Decontamination Equipment	Centrifuge Transporter[3]	3
	Cranes (RMC)[3]	8
	Cranes, Bridge X-7725[3]	2
	Centrifuge Mobile Equipment[3]	4
	Centrifuge Dismantling Equipment (X-7725 Assembly Stands)	6

[1] Includes 11,520 operational units plus contaminated spare centrifuges.
[2] Wall surface areas excluded since these areas are not anticipated to require decontamination.
[3] Equipment re-utilized from operational phase.

Components	Description [units]	Estimated Quantity
Decontamination Equipment (Continued)	Cutting Machines	2
	Degreasers	2
	Decontamination Tanks	4
	Wet Blast Cabinets	2
	Crusher	1

10.3 Management/Organization

Management of the decommissioning program will assure proper training and procedures are provided to assure worker health and safety. The programs will focus on minimizing waste volumes and worker exposure to hazardous or radioactive materials. Qualified contractors assisting with decommissioning will be subject to ACP security and training requirements, and procedural controls.

10.4 Health and Safety

Consistent with the policy during ACP operation, the policy during decommissioning is to keep individual and collective occupational radiation exposure with the ALARA principle. A Radiation Protection Program will identify and control sources of radiation, establish worker protection requirements and direct the use of survey and monitoring instruments.

10.5 Waste Management

Radioactive and hazardous wastes produced during decommissioning will be collected, handled, and disposed of in accordance with regulations applicable to the ACP at the time of decommissioning. Generally, procedures will be similar to those described for wastes produced during operation. These wastes will ultimately be disposed of in licensed radioactive or hazardous waste disposal facilities. Non-hazardous and non-radioactive wastes will be disposed of consistent with good industrial practice, and in accordance with applicable regulations.

10.6 Security and Nuclear Material Control

Requirements for physical security and for nuclear material control and accountability will be maintained during decommissioning in a manner similar to the programs in force during ACP operation. This includes requirements for control of classified information and classified

liquid waste from the centrifuge internals and 1,730,000 cubic feet of classified waste in non-reusable packaging.

Equipment and Supply: $15 million

This includes the purchase or lease of dismantling, cutting, degreasing, and crushing equipment; decontamination tanks, wet blast cabinets, and over 20,000 containers (B-25 boxes and 55 gallon drums).

Laboratory: $1.3 million

This includes labor costs for sampling, transport, testing, and analysis of samples.

Indirect Services: $33.6 million

This includes support services (such as laundry, janitorial, etc) and infrastructure costs (such as water, power, etc) not included in other tasks.

Miscellaneous: $27.6 million

This includes direct costs of $2.5 million for miscellaneous material for decommissioning and $25.1 million for indirect costs, such as NRC review fees for the submitted DP, license fees, DOE lease fees, business insurance, and taxes.

Subtotal	**$171.3 million**
General and Administrative (6 percent)	**$10.3 million**
Contractor Profit (15 percent)[4]	**$23.5 million**
Contingency (25 percent)	**$51.3 million**
Total Plant Decommissioning Cost Estimate	**$256.4 million**

[4] Contractor Profit = 0.15[(Subtotal + General and Administrative) - (NRC Review Fees + License Fees + DOE Lease Fees)]

10.10.2 UF_6 Tails Disposition Costs

Cost estimates to dispose of UF_6 tails generated during ACP operation are separate from the cost estimates to decommission the plant. As noted previously, the ultimate disposal of UF_6 tails remains to be determined. USEC intends to evaluate possible commercial uses of UF_6 tails before having the tails processed by the DOE UF_6 conversion facility in Piketon, Ohio. UF_6 tails are stored in steel cylinders until they can be processed in accordance with the disposal strategy established by USEC. Depending on technological developments and the existence of facilities available prior to ACP shutdown, the tails may have commercial value and may be marketable for further enrichment or other processes. However, for the purposes of calculating the UF_6 tails disposition cost, USEC assumes that the total quantity of tails generated during ACP operation are processed by the DOE UF_6 conversion facility in Piketon, Ohio.

For conservatism, USEC provides financial assurance to fund the estimated cost of conversion and disposal of the depleted uranium inventory as it is generated during ACP operation. This funding is described in the DFP and is in addition to the funding requirements for decommissioning the ACP. As with plant decommissioning, the cost estimate will likely change between the time of license issuance and actual decommissioning. USEC commits to adjust the cost estimate for tails disposal prior to operation of each additional increment of capacity on process gas and no less frequent than annually, once full capacity is achieved. The method for adjusting the cost estimate will consider the same factors as previously described in Section 10.10.1 of this chapter.

At full capacity, the ACP will generate approximately 9,520 MT of UF_6 tails annually. As with other decommissioning costs, the disposal cost estimate for UF_6 tails disposal is provided in 2004 dollars. In view of the commitment to annually adjust tails disposal cost estimates, the ability to know with certainty the tails inventory from prior years of ACP operation, and USEC's demonstrated ability to accurately and conservatively predict anticipated tails generation one year ahead of time, a 10 percent contingency factor is applied to the tails disposal cost estimate. This contingency factor is consistent with that used for tails generated from the United States Enrichment Corporation's GDP operations. The total estimated cost to dispose of UF_6 tails over the 30-year license, including a four-year ramp up to full capacity and the 10 percent contingency factor, is \$591.9 million. The basis for this estimate is provided in the DFP.

10.10.3 Total Decommissioning Liability

USEC's total decommissioning liability is the sum of the total plant decommissioning costs and the tails disposition costs. USEC's total liability for decommissioning the ACP, including applicable contingencies, is:

Plant Decommissioning Cost	\$256.4 million
UF_6 Tails Disposition Cost	\$591.9 million
Total Decommissioning Liability	\$848.3 million

Table 10.10-1 Plant Decommissioning Cost Estimates and Expected Duration

Task/Item	Cost Estimate (Millions, 2004 dollars)	Approx Percentage
Planning and Preparation	$2.6	2%
Decontamination and/or Dismantling of Radioactive Facilities	$39.6	23%
Restoration of Contaminated Areas On Plant Grounds	$0.7	0%
Final Status Survey	$1.0	1%
Site Stabilization and Long-Term Surveillance	$2.4	1%
Packing Materials, Shipping, and Waste Disposal	$47.5	28%
Equipment and Supply	$15.0	9%
Laboratory	$1.3	1%
Indirect Services	$33.6	20%
Miscellaneous	$27.6	15%
Subtotal	$171.3	100%
General and Administrative (6%)	10.3	
Contractor Profit (15%)	23.5	
Contingency (25%)	$51.3	
Total Plant Decommissioning Cost	$256.4	
UF_6 Tails Disposal Costs UF_6 Tails Contingency (10%)	$538.1 53.8	
Total UF6 Tails Disposition Cost	$591.9	
Total Decommissioning Liability	$848.3	

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NR-3605-0006

Decommissioning Funding Plan

for the American Centrifuge Plant

in Piketon, Ohio

The American Centrifuge™

USEC's Advanced Uranium Enrichment Technology

Revision 2

Docket No. 70-7004

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May 2005

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Date: 05/23/05

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NR-3605-0006

DECOMMISSIONING FUNDING PLAN
for the American Centrifuge Plant
in Piketon, Ohio

Docket No. 70-7004

Revision 2

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UPDATED LIST OF EFFECTIVE PAGES

Revision 0 – 10 CFR 1045 review completed by L. Sparks on 07/29/04 and the Export Controlled Information review completed by R. Coriell on 07/30/04.

Revision 1 – 10 CFR 1045 review completed by L. Sparks on 03/07/05 and the Export Controlled Information review completed by R. Coriell on 03/10/05.

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Table C3.7 Decontamination or Dismantling of Radioactive Facility Components (Productive Work Days)

Group		Type	# Workers	Dur (#y)	Prod Factor	Total (wd)
Supervision		Salary	6	5	219	6,570
Engineering		Salary	5	5	219	5,475
Operations		Salary	3	5	219	3,285
		Hourly	21	5	219	22,995
Maintenance		Salary	9	5	219	9,855
		Hourly	44	5	219	48,180
Support	Plant Support	Salary	5	5	219	5,475
		Hourly	17	5	219	18,615
	Production Support	Salary	8	5	219	8,760
		Hourly	11	5	219	12,045
Total			129			141,255

Assumptions:
Anticipated duration = 5 y
Productivity Factor = 219 wd/y = 260 - 41 (Paid Absences)
Constant $ Pay
Security - 4 Stations - manned (typical) 24/7

Anticipated tasks considered:
Erect Decontamination Facility
Decontamination of facilities - Internals
Dismantle centrifuge machines; Waste segregation
Dismantle facilities/components
Tails Cylinder movement/disposition [material title transfer DOE/UDS]
Continued Project and Security Support

Table C3.8 Restoration of Contaminated Areas on Facility Grounds (Productive Work Days)

Group		Type	# Workers	Dur (#y)	Prod Factor	Total (wd)
Supervision		Salary	0	2	219	0
Engineering		Salary	0	2	219	0
Operations		Salary	1	2	219	438
		Hourly	5	2	219	2,190
Maintenance		Salary	0	2	219	0
		Hourly	0	2	219	0
Support	Plant Support	Salary	1	2	219	438
		Hourly	0	2	219	0
	Production Support	Salary	0	2	219	0
		Hourly	0	2	219	0
Total			7			3,066

Assumptions:
Anticipated duration = 2y
Productivity Factor = 219 wd/y = 260 - 41 (Paid Absences)
Constant $ Pay
1 person cleans ~600 - 900 ft^2/d (750 ft^2/d used) loose contamination (minimal amount of loose contamination anticipated)
Shares resource allocation coincident with Decontamination or Dismantling phase effort
Minimal loose contamination and cleanup anticipated
Labor estimate includes non-labor cots for analytical sampling/surveying efforts.

Anticipated tasks considered:
Decontamination of facilities - external/outside; cylinder yards
Perform HP surveys
Remove fixed contamination; Scarify cylinder storage yards surfaces
Collect/dispose of yard debris

Table C3.9 Final Radiation Survey (Productive Work Days)

Group		Type	# Workers	Dur (#y)	Prod Factor	Total (wd)
Supervision		Salary	0	2.5	219	0
Engineering		Salary	1	2.5	219	548
Operations		Salary	0	2.5	219	0
		Hourly	0	2.5	219	0
Maintenance		Salary	0	2.5	219	0
		Hourly	1	2.5	219	548
Support	Plant Support	Salary	3	2.5	219	1,643
		Hourly	1	2.5	219	548
	Production Support	Salary	0	2.5	219	0
		Hourly	0	2.5	219	0
Total			6			3,285

<u>Assumptions</u>:
Anticipated duration = 2.5y
Productivity Factor = 219 wd/y = 260 - 41 (Paid Absences)
Constant $ Pay
Work period occurs coincident with the last 2.5 years of the D&D phase.
Labor estimate includes non-labor cots for analytical sampling/surveying efforts.

<u>Anticipated tasks considered</u>:
Develop/implement survey plans
Collect/analyze data
Perform confirmatory surveys
Develop final survey report
Terminate license

Table C3.10 Site Stabilization and Long-Term Surveillance (Productive Work Days)

Group		Type	# Workers	Dur (#y)	Prod Factor	Total (wd)
Supervision		Salary	0	6	219	0
Engineering		Salary	1	6	219	1,314
Operations		Salary	1	6	219	1,314
		Hourly	1	6	219	1,314
Maintenance		Salary	0	6	219	0
		Hourly	2	6	219	2,628
Support	Plant Support	Salary	0	6	219	0
		Hourly	0	6	219	0
	Production Support	Salary	0	6	219	0
		Hourly	0	6	219	0
Total			5			6,570

<u>Assumptions</u>:
Anticipated duration = 6y (coincident with P&P and D&D)
Productivity Factor = 219 wd/y = 260 - 41 (Paid Absences)
Constant $ Pay

<u>Anticipated tasks considered</u>:
Site stabilization - not required
Maintain maintenance/surveillances on IROFS equipment necessary until license terminated (~ year six)

Table C3.19 Estimated Volume of Annual Depleted Uranium Generated

Calendar Year	[Q] # Machines		[R] DUF_6 Generated [1,000 MT]	[S] DUF_6 Accumulated [1,000 MT]	[T] DU Accumulated [1,000 MT]	[U] Tails Disposal Cost [$, 2004]	[V] # Tails Cylinders
2006	200		0	0	0	$0	0
2007	120	*	0.099	0.099	0.067	$201,070	8
2008	2700		2.23	2.33	1.51	$4,524,071	179
2009	7300		6.03	8.36	4.08	$12,231,748	483
2010	11520		9.52	17.88	6.43	$19,302,703	763
2011-2036	11520		247.43	265.30	167.29	$501,870,283	19,836
Total			265.30	265.30	179.38	**$538,129,875**	21,269

*** - based upon Lead Cascade potential Production capabilities that can produce material & number of machines considered.**

<u>Assumptions</u>:

Operational (license) life = 30 years (from 2006 – 2036); 365 days/yr; 24 hr/day
Tails Output during Operation (@ 3,500 MTSWU/yr) = 2,395 lbs. UF_6/hr
Weight Conversion Factor = 0.45359 Kg/lb; Tails Material Conversion Factor = 0.30668 Kg/lb UF_6; Tails Purity = 0.67612 gU/g; based upon 0.35% Average Tails
U disposal cost = $3/Kg U
R=Q/11,520*number of years*2,395*24*365; T=R*0.67612; U=T*3
V=R*1,000,000/0.45359/27,500
~21,269 Tails cylinders generated; 27,500 # UF_6 fill weight = 1,000 generated parent cylinders (@ EOL)

Table C3.20 Total Labor Distribution

Group		Type	Job/Personnel Descriptions
Supervision		Salary	Program Manager, Project Manager, Office Manager, QA/Reg Manager, Rad-Environmental-Safety and Health Manager, FNMCA Manager
Engineering		Salary	Design Engineer, Field Support, NCS Engineer, Nuclear Safety, Regulatory
Operations		Salary	Operations FLM
		Hourly	Chemical Operations, UMH
Maintenance		Salary	Maintenance FLM, Scheduler-Planner
		Hourly	Mechanic, Laborer, Field Service Technician
Support	Plant Support	Salary	HP Support
		Hourly	Protection Forces
	Production Support	Salary	Waste Engineer
		Hourly	Waste Handler

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