UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
USEC Inc.	)	Docket No. 70-7004
	)	
(American Centrifuge Plant)	)	ASLBP No. 05-838-01-ML

NRC STAFF RESPONSE TO ATOMIC SAFETY AND LICENSING BOARD ORDER OF FEBRUARY 6, 2007

Margaret J. Bupp
Brett M. Klukan
Counsel for the NRC Staff

February 20, 2007

TABLE OF CONTENTS

February 20, 2007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
USEC Inc.	)	Docket No. 70-7004
	)	
(American Centrifuge Plant)	)	ASLBP No. 05-838-01-ML

NRC STAFF RESPONSE TO ATOMIC SAFETY AND LICENSING BOARD ORDER OF FEBRUARY 6, 2007

INTRODUCTION

On February 6, 2007, the Atomic Safety and Licensing Board ("Board") issued an order modifying the case schedule in the above-captioned proceeding and issuing questions to be answered by the NRC staff ("Staff"), as well as identifying topics for the upcoming uncontested licensing hearing.[1] Pursuant to the deadline established in the February 6 Order, the Staff hereby responds to the Board's questions. As directed in the February 6 Order, the Staff has, to the extent possible, coordinated its responses with the license applicant, USEC, Inc. ("the Applicant").

BACKGROUND

On August 23, 2004, the Applicant filed an application to construct and operate a gas centrifuge uranium enrichment facility, to be known as the American Centrifuge Plant (ACP), on land leased from the Department of Energy (DOE) in Piketon, Ohio. The Staff published a "Notice of Receipt of Application for a License; Notice of Availability of Applicant's Environmental Report; Notice of Consideration of Issuance of License; and Notice of Hearing and Commission Order" ("Notice of Hearing") in the *Federal Register* on October 18, 2004.

---

[1] Order (Establishing a Modified Case Schedule)(Issuing Questions and Identifying Hearing Topics), Feb. 6, 2007. ("February 6 Order")

69 Fed. Reg. 61411 (Oct. 18, 2004). In the Notice of Hearing, the Commission directed the Board to conduct a hearing in accordance with 10 C.F.R. Part 2 and to make certain findings required by 10 C.F.R. § 2.104(b), a so-called “mandatory hearing.”

Two petitions to intervene were filed pursuant to 10 C.F.R. § 2.309. The Commission found that both petitioners, Portsmouth/Piketon Residents for Environmental Safety and Security (“PRESS”) and individual Geoffrey Sea, had standing to intervene. *USEC Inc.* (ACP), CLI-05-11, 61 NRC 309, 310 (2005). However, the Board ultimately found that neither PRESS nor Mr. Sea had proffered an admissible contention. *USEC, Inc.* (ACP), LBP-05-28, 62 NRC 585 (2005); *aff’d ACP*, CLI-06-09, 63 NRC 451 (2006) (Sea Contentions); *aff’d USEC Inc.* (ACP), CLI-06-10, 63 NRC 433 (2006) (PRESS Contentions).

In order to proceed with the mandatory hearing, the Board requested documents related to the Staff’s review of the license application (“the Application”). Order (Request for Documents and Briefings), April 19, 2006; Memorandum and Order (Ruling on Motion for Modification and Clarification), May 31, 2006. The Board requested, and the Staff and USEC provided, copies of the Application, including the Emergency Plan (EP), Integrated Safety Analysis (ISA) calculations previously submitted to the Staff, the Physical Security Plan, the Fundamental Nuclear Material Control Plan, and the Environmental Report (ER); the ISA Summary; the Staff’s final Environmental Impact Statement (FEIS), NUREG-1834, “Environmental Impact Statement for the Proposed American Centrifuge Plant in Piketon, Ohio,” Final Report (2006); the Staff’s Safety Evaluation Report (SER), NUREG-1851, “Safety Evaluation Report for the American Centrifuge Plant in Piketon, Ohio” (2006); Staff Requests for Additional Information (RAIs) and USEC Responses; and information related to presentations to the Advisory Committee on Nuclear Waste (ACNW) related to the Application.

In addition to the above documentation related to the Application, the Board also requested that the Staff, after completing the SER and FEIS and submitting them to the Board, provide proposed findings of fact and conclusions of law based upon the Staff's review, as outlined in the SER and FEIS.[2] Together with the Staff Findings, the SER and the FEIS represent the Staff's conclusions based on its review of the Application and associated documents from the Applicant. Following the issuance of the Board's final initial decision and the grant of the license (assuming the Agency decision is to issue a license), construction of the facility may begin. In accordance with 10 C.F.R. § 70.72(d)(2), the Applicant (then Licensee) will submit to the Staff annual updates to the ISA Summary during construction along with a brief summary of any changes to the facility design made during the year. In addition, the Applicant has committed to provide to the Staff an update to the ISA Summary at least 180 days prior to the planned introduction of special nuclear material into the ACP facility. The Staff will review these submissions as well as any license amendment requests that may be submitted. Although the Applicant (then Licensee) can start construction following issuance of the license, it may not begin operation of the enrichment facility until after it successfully completes a second step. Prior to operation, the Staff must verify through inspection that the facility has been constructed in accordance with the requirements of the license pursuant to 10 C.F.R. § 70.32(k). Only after this second step is successfully completed will the ACP be permitted to begin operations. SER at xvii.

---

[2] "NRC Staff's Proposed Findings of Fact and Conclusions of Law in the Mandatory Hearing," October 11, 2006. ("Staff Findings")

In response to the Staff Findings, the Applicant submitted comments.[3] After the receipt of documentation related to the Application, the Staff Findings, and the USEC Findings, the Board set a tentative schedule for the remainder of the mandatory hearing.[4] The Scheduling Order included dates for a limited appearance session and a visit by the Board and representatives of the Staff and the Applicant to the proposed ACP site in Piketon, Ohio. Those events took place on January 9 and 10, 2007, respectively. The Scheduling Order stated that the Board would identify issues for the mandatory hearing and related questions to the Staff on February 13, 2007, and that the Staff would file pre-filed testimony on the hearing issues by March 12, 2007. A pre-hearing conference was scheduled for April 9, 2007, with the substantive portion of the mandatory hearing to begin on April 10, 2007. In response to the Scheduling Order, the Applicant filed a motion requesting that the Board accelerate the mandatory hearing schedule.[5] The Board denied the Motion,[6] finding that the original schedule was very ambitious, and that the Board would not have sufficient time to conduct a thorough review of the licensing under the proposed accelerated schedule.

On February 1, 2007, the Commission issued an Order directing the Board to modify the tentative case schedule established under the Scheduling Order.[7] Pursuant to the

---

[3] "USEC Inc.'s Comments on NRC Staff's Proposed Findings of Fact and Conclusions of Law in the Mandatory Hearing," October 19, 2006. ("USEC Findings")

[4] Order (Establishing Tentative Case Schedule), November 17, 2006. ("Scheduling Order")

[5] "USEC Inc. Motion to Accelerate Mandatory Hearing Schedule and For Opportunity to Submit Prefiled Direct Testimony," November 21, 2006 ("Motion").

[6] Memorandum and Order (Denying USEC's Motion to Accelerate Mandatory Hearing Schedule and Establishing Guidelines for the Submission of Pre-filed Testimony by the Applicant), December 22, 2006.

[7] *USEC, Inc.* (American Centrifuge Plant), CLI-07-05, 65 NRC __ (slip op.) (Feb. 1, 2006) ("February 1 Order").

February 1 Order, the Board issued the February 6 Order establishing new deadlines for the upcoming steps in the mandatory process and issuing questions and identifying hearing topics.

RESPONSE TO BOARD'S QUESTIONS

Each of the Board's questions is restated below, followed by the Staff's response.[8] In instances where the Staff's response contains information designated Official Use Only-DOE NOFORN, the response has been submitted in a separate appendix (Appendix A) that is being withheld from public disclosure.

A. SAFETY ISSUES

G1. Review Process

The NRC Staff has developed generic guidance for reviewing applications for licenses for fuel cycle facilities, including enrichment and fuel fabrication facilities. In regards to the general review procedures, the Board directs the Staff to address the following:

For each review area presented in NUREG-1520:

A. Summarize:
   1. Specific cases where the areas of review, acceptance criteria, and review procedures were not followed by the Staff.
   2. Rationale for why NUREG-1520 was not followed in each of those cases.
   3. Alternative procedures that were used in its review and provide a justification as to why the alternative procedures are equivalent or superior to those presented in NUREG-1520.

Staff Response: For the sake of clarity and efficient organization, the Staff structures its response to this question by SER chapter. For each SER chapter, any instances in which the Staff used a different approach from that suggested in NUREG-1520 are identified and explained.

[8] The Staff's responses are supported by affidavits from the relevant subject matter experts who provided the technical information for each response. (Attached). Statements of the Staff's subject matter experts' professional qualifications will be included with the Staff's testimony, as directed by the February 6 Order.

SER CHAPTER 2.0 "ORGANIZATION AND ADMINISTRATION."

The guidance applicable to the NRC's review of the organization and administration section of the Application is contained in Chapter 2 of NUREG-1520 (SER at 2-1), subject to the following exceptions:

1. NUREG-1520 Section 2.3:

Section 2.3 of NUREG-1520, "Areas of Review," applies a set of acceptance criteria to new facilities distinct from that which it applies to modifications of existing facilities. Since, as stated in the SER, "the American Centrifuge Plant (ACP) is a new facility, the areas of review for existing facilities are not applicable." *Id.*

2. NUREG-1520 Section 2.4.3:

Similar to NUREG-1520 Section 2.3, NUREG-1520 Section 2.4.3, "Regulatory Acceptance Criteria," enumerates acceptance criteria for two categories of facilities, "New Facilities or Facilities Undergoing Major Modification," and "Existing Facilities." As the ACP would be a new facility, only the acceptance criteria for new facilities in NUREG-1520 Section 2.4.3 apply. *Id.*

SER CHAPTER 3.0 "INTEGRATED SAFETY ANALYSIS (ISA AND ISA SUMMARY)."

The guidance applicable to the NRC's review of the Applicant's ISA and ISA Summary is contained in Chapter 3 of NUREG-1520. SER at 3-3; *see also* Staff Findings at 16. Chapter 3 is applicable in its entirety (*id.*), with the following three exceptions:

1. NUREG-1520 Section 3.4.3.2(4)(c):

As the subject of Section 3.4.3.2(4)(c), criticality monitoring, is already addressed as part of Chapter 5 of the SER, there was no need to address that subject again as part of SER Chapter 3. SER at 3-3; *see also* Staff Findings at 16.

2. NUREG-1520 Section 3.4.3.2(5)(b)(i-ix):

NUREG-1520 Section 3.4.3.2(5)(b)(i-ix), in part, provides:

> Process Hazard Analysis Method. The process hazard analysis method is acceptable *if it involves selecting one of the methods described in NUREG-1513 in accordance with the selection criteria established in that document.* Methods not described in NUREG-1513 may be acceptable provided that they fulfill the following conditions…

(Emphasis added). "Because the methods used by the Applicant (preliminary hazard analysis method and the what if/checklist method (WI/CL)) are described in NUREG-1513 (NRC, 2001), these [foregoing] conditions in Section 3.4.3.2(5)(b)(i-ix) d[id] not have to be addressed." SER at 3-3; *see also* Staff Findings at 16. Thus, per the Applicant's selection of a hazard analysis method as described in NUREG-1513, the Staff's review was based on the acceptance criteria found in that document, instead of those in NUREG-1520 Section 3.4.3.2(5)(b)(i-ix). *Id.*

3. NUREG-1520 Section 3.4.3.2(9):

NUREG-1520 Section 3.4.3.2(9) supplies guidance for the descriptive composition and conduct of qualitative methodologies as employed in event-likelihood analysis. Since the Applicant chose instead to employ a quantitative analysis, as NUREG-1520 Sections 3.4.3.1 and 3.4.3.2 permit, as opposed to a qualitative analysis, the acceptance criteria for qualitative descriptors in NUREG-1520 Section 3.4.3.2(9) did not have to be addressed. *Id.* Instead, as the SER states, the applicable acceptance criteria for quantitative methodology employed by the Applicant are contained in NUREG-1520 Sections 3.4.3.1 and 3.4.3.2. *Id.*

SER CHAPTER 6.0 "CHEMICAL PROCESS SAFETY."

The guidance applicable to the Staff's review of chemical process safety for the ACP is contained in Chapter 6 of NUREG-1520. SER at 6-1; *see also* Staff Findings at 34.

This chapter is applicable in its entirety. *Id.* The Staff also relied upon the following as additional guidance:

1. NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities" (1997).
2. NUREG-1513, "Integrated Safety Analysis Guidance Document" (2001). *Id.*

SER CHAPTER 8.0 "EMERGENCY MANAGEMENT."

The guidance applicable to the Staff's review of the emergency management plan in the Application is contained in Chapter 8 of NUREG-1520, subject to the following exceptions (SER at 8-1):

1. NUREG-1520 Section 8.3.2:

NUREG-1520 Section 8.3.2, "Evaluation That No Emergency Plan Is Required," provides

> If the Applicant submits an evaluation or references the ISA Summary *to demonstrate that an emergency plan is not required*, the Staff should review the information against 10 CFR 70.22(i)(1)(i), and NUREG-1140, "A Regulatory Analysis of Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees." NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," dated March 1998, also contains useful information. Areas evaluated should include the following:
> (1) a description of the facility
> (2) types of materials used, including both radioactive material and hazardous chemicals
> (3) types of accidents
> (4) detection of accidents
> (5) site specific information used to support the evaluation
> (6) an evaluation of the consequences.

(Emphasis added). As the Applicant did submit an emergency plan as part of its Application, NUREG-1520 Section 8.3.2 is inapplicable. *Id.* The Staff reviewed the EP in the Application as per the acceptance criteria in Section 8.4.3.1. *Id.*; *see also* Staff Findings at 39.

2. NUREG-1520 Section 8.6.2.2:

When an applicant asserts that an emergency plan is not necessary, NUREG-1520 Section 8.6.2.2 dictates the review process by which the Staff evaluates the validity of that assertion. As the Application submitted by USEC included an EP, NUREG-1520 Section

8.6.2.2 is not applicable. SER at 8-1. As part of the SER, the Staff reviewed the EP in the Application as per the acceptance criteria in Section 8.4.3.1. *Id.*; *see also* Staff Findings at 39.

SER CHAPTER 9.0 "ENVIRONMENTAL PROTECTION."

With the exception of the acceptance criteria in NUREG-1520 Section 9.4.3.2, "Environmental Protection Measures," the acceptance criteria found in NUREG-1520 Section 9.4 are not applicable. SER at 9-1; *see also* Staff Findings at 49. Those non-applicable provisions provide guidance to the Staff for determining whether the Staff should prepare an environmental assessment (EA) or an environmental impact statement (EIS). SER at 9-1. The Staff has already prepared and finalized an EIS in accordance with 10 C.F.R. Part 51, thus rendering these provisions moot. *Id.* Section 51.20(b)(10) requires the preparation of an EIS with respect to any proposed uranium enrichment facility.

SER CHAPTER 10.0 "DECOMMISSIONING."

The guidance applicable to the Staff review of the decommissioning section of the Application is contained in Chapter 10 of NUREG-1520 and in Volume 3 of NUREG-1757, "Consolidated NMSS Decommissioning Guidance" (2003). SER at 10-2. NUREG-1757 is the updated version of NUREG-1727, "NMSS Decommissioning Standard Review Plan" (2000), referenced in NUREG-1520. *Id.*; *see also* Staff Findings at 54-55.

SER APPENDIX A "INTEGRATED SAFETY ANALYSIS (ISA) AND ISA SUMMARY."

The guidance applicable to the Staff's review of the Applicant's ISA and ISA Summary is contained in Chapter 3 of NUREG-1520. SER at A-2; *see also* Staff Findings at 68. Chapter 3 is applicable in its entirety, with the following three exceptions (*Id.)*:

1. NUREG-1520 Section 3.4.3.2(4)(c):

As the subject of Section 3.4.3.2(4)(c), criticality monitoring, is already addressed as part of Chapter 5 of the SER, there was no need to address that subject again as part of SER Appendix A. *Id.*

2. NUREG-1520 Section 3.4.3.2(5)(b)(i-ix):

NUREG-1520 Section 3.4.3.2(5)(b)(i-ix), in part, provides:

> Process Hazard Analysis Method. The process hazard analysis method is acceptable *if it involves selecting one of the methods described in NUREG-1513 in accordance with the selection criteria established in that document.* Methods not described in NUREG-1513 may be acceptable provided that they fulfill the following conditions…

(Emphasis added). "Because the methods used by the Applicant (preliminary hazard analysis method and the what if/checklist method (WI/CL)) are described in NUREG-1513 (NRC, 2001), these [foregoing] conditions in Section 3.4.3.2(5)(b)(i-ix) d[id] not have to be addressed." SER at A-2; *see also* Staff Findings at 68. Thus, per the Applicant's selection of a hazard analysis method as described in NUREG-1513, the Staff reviewed the acceptance criteria found in that document, instead of those in NUREG-1520 Section 3.4.3.2(5)(b)(i-ix). *Id.*

3. NUREG-1520 Section 3.4.3.2(9)

NUREG-1520 Section 3.4.3.2(9) supplies guidance for the descriptive composition and conduct of qualitative methodologies as employed in event-likelihood analysis. Since the Applicant chose instead to employ a quantitative analysis, as NUREG-1520 Sections 3.4.3.1 and 3.4.3.2 permit, as opposed to a qualitative analysis, the acceptance criteria for qualitative descriptors in NUREG-1520 Section 3.4.3.2(9) did not have to be addressed. *Id.* Instead, as the SER states, the applicable acceptance criteria for quantitative methodology employed by the Applicant are contained in NUREG-1520 Sections 3.4.3.1 and 3.4.3.2. *Id.*

SER APPENDIX B "ACCIDENT ANALYSIS FOR THE PROPOSED ACP."

As NUREG-1520 does not apply to this section of the SER (a summary of the Staff's independent evaluation of the consequences of a selected representative set of potential accidents), NUREG-1520 was not considered. *See* SER at B-1.

SER APPENDIX G "HUMAN FACTORS."

The guidance applicable to the Staff's review of the human factors engineering description portions of the Application and ISA Summary is found in:

1. Chapter 18, "Human Factors Engineering," of NUREG-0800 (2004),
2. NUREG-700, Human System Interface Design Review Guideline" (2002), and
3. NUREG-0711, "Human Factors Engineering Program Review Model" (2004).

SER at G-1; *see also* Staff Findings at 78. As NUREG-1520 does not apply to the foregoing area of review, it was not considered. *See id.*

SER APPENDIX H "MATERIAL CONTROL AND ACCOUNTING."

The guidance applicable to the Staff's review of the material control and accounting description in the Fundamental Nuclear Material Control Plan is found in:

1. NUREG/CR-5734, "Recommendations to the NRC on Acceptable Standard Format and Content for the Fundamental Nuclear Material Control (FNMC) Plan Required for Low-Enriched Uranium Enrichment Facilities" (1991);
2. Regulatory Guide 5.67, "Material Control and Accounting Requirements for Uranium Enrichment Facilities Authorized to Produce Special Nuclear Material of Low Strategic Significance" (1993);
3. NUREG/BR-0096, "Instructions and Guidance for Completing Physical Inventory Summary Reports" (1992);
4. NUREG/BR-0006, "Instructions for Completing Nuclear Material Transaction Reports" (2003); and
5. NUREG/BR-0007, "Instructions for the Preparation and Distribution of Material Status Reports" (2003).

SER at H-1; *see also* Staff Findings at 79-80. As NUREG-1520 does not apply to the foregoing area of review, it was not considered. *See id.*

<u>SER APPENDIX I "PHYSICAL PROTECTION."</u>

The guidance applicable to the Staff's review of physical protection found in the physical security plan is contained in Part II, "Special Nuclear Material of Low Strategic Significance," of Regulatory Guide 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear material of Moderate to Low Strategic Significance" (1982). SER at I-1 to I-2; *see also* Staff Findings at 81. As NUREG-1520 does not apply to the foregoing area of review, it was not considered. *See id.*

<u>SER APPENDIX J "PHYSICAL SECURITY OF THE TRANSPORTATION OF SPECIAL NUCLEAR MATERIAL OF LOW STRATEGIC SIGNIFICANCE."</u>

As NUREG-1520 does not apply to the foregoing area of review, it was not considered. *See* SER at J-1; *see also* Findings at 82-83.

B. For any areas where the Staff used other guidance, provide:
1. A list of the guidance document(s).
2. A list of the topics that were reviewed by the document(s).
3. Notification as to whether the other guidance documents supplemented or replaced the review procedures in NUREG-1520.

Staff Response: The Staff provides its response to this part per the following table:

<u>US NRC NUREG-0700, Revision 2 "Human-System Interface Design Review Guidance" (2002).</u>

<u>SER Section Citing to Document.</u>	**<u>Topic Addressed / Discussed</u>**	**<u>Role / Purpose of Document in Relation to NUREG-1520</u>**
SER Appendix G, "Human Factors."	Human-System Interface Design Review.	**Supplementation of NUREG-1520.** Whether acceptable human factors engineering practices and guidance were incorporated into the facility's design basis.

US NRC NUREG-0711, "Human Factors Engineering Program Review Model," (2004).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Appendix G, "Human Factors."	Human Factors Engineering Program Review.	**Supplementation of NUREG-1520.** Whether acceptable human factors engineering practices and guidance were incorporated into the facility's design basis.

US NRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (2004).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Appendix G, "Human Factors."	NUREG-0800 Ch. 18, "Human Factors Engineering."	**Supplementation of NUREG-1520.** Whether acceptable human factors engineering practices and guidance were incorporated into the facility's design basis.

US NRC NUREG-1513 "Integrated Safety Analysis Guidance Document" (2001).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Chapter 3, "Integrated Safety Analysis (ISA) and ISA Summary."	Section of Hazard Evaluation Method.	**Replacement of NUREG-1520 Section 3.4.3.2(5)(b)(i-ix).** That Section of NUREG-1520 provides acceptance criteria for hazard analysis methods only if the proposed methods are not otherwise described in NUREG-1513.

US NRC NUREG-1513 “Integrated Safety Analysis Guidance Document" (2001).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Chapter 3, "Integrated Safety Analysis (ISA) and ISA Summary."	Section of Hazard Evaluation Method.	**Replacement of NUREG-1520 Section 3.4.3.2(5)(b)(i-ix).** That Section of NUREG-1520 provides acceptance criteria for hazard analysis methods only if the proposed methods are not otherwise described in NUREG-1513.
SER Section Citing to Document.	**Topic Addressed / Discussed**	**Role / Purpose of Document in Relation to NUREG-1520**
SER Chapter 6, "Chemical Process Safety."	ISA and ISA Summary Review.	Supplementation of NUREG-1520.
SER Section Citing to Document.	**Topic Addressed / Discussed**	**Role / Purpose of Document in Relation to NUREG-1520**
SER Appendix A "Integrated Safety Analysis (ISA) and ISA Summary."	Section of Hazard Evaluation Method.	**Replacement - NUREG-1520 Section 3.4.3.2(5)(b)(i-ix).** That Section of NUREG-1520 provides acceptance criteria for hazard analysis methods only if the proposed methods are not otherwise described in NUREG-1513.

US NRC NUREG-1601 “Chemical Process Safety at Fuel Cycle Facilities" (1997).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Chapter 6, "Chemical Process Safety."	Chemical Process Safety at Fuel Cycle Facilities.	**Supplementation of NUREG-1520.**

US NRC NUREG-1757 "Consolidated NMSS Decommissioning Guidance," September, (2003).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Chapter 10, "Decommissioning."	Overview of Proposed Decommissioning Activities: 1.) Conceptual Decommissioning and Decontamination Planning and 2.) Decommissioning Costs and Financial Assurance.	**Supplementation of NUREG-1520 Section 10.1** Section 10.1 of NUREG-1520 provides what overview of proposed decommissioning activities is necessary at the time of initial licensing or license renewal. NUREG-1757 supplements that generalized discussion with a more detailed discussion of the relevant information requirements.

US NRC NUREG/BR-0006, Revision 6 "Instructions for Completing Nuclear Material Transaction Reports" (2003).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Appendix H, "Material Control and Accounting."	Material Control and Accounting Description Review.	**Replacement of NUREG-1520.** NUREG-1520 does not apply to this part.

US NRC NUREG/BR-0007, Revision 5 "Instructions for the Preparation and Distribution of Material Status Reports" (2003).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Appendix H, "Material Control and Accounting."	Material Control and Accounting Description Review.	**Replacement of NUREG-1520.** NUREG-1520 does not apply to this part.

US NRC NUREG/BR-0096 "Instructions and Guidance for Completing Physical Inventory Summary Reports" (1992).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Appendix H, "Material Control and Accounting."	Material Control and Accounting Description Review.	**Replacement of NUREG-1520.** NUREG-1520 does not apply to this part.

US NRC NUREG/CR-5734 "Recommendations to the NRC on Acceptable Standard Format and Content for the Fundamental Nuclear Material Control Plan Required for Low-Enriched Uranium Enrichment" (1991).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Appendix H, "Material Control and Accounting."	Material Control and Accounting Description Review.	**Replacement of NUREG-1520.** NUREG-1520 does not apply to this part.

US NRC Regulatory Guide 5.59 "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate to Low Strategic Significance" (1982).

SER Section Citing to Document.	Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Appendix I, "Physical Protection."	Regulatory Guide 5.59 Part II, "Special Nuclear Material of Low Strategic Significance."	**Replacement of NUREG-1520.** NUREG-1520 does not apply to this part.

SER Section Citing to Document.	US NRC Regulatory Guide 5.67 "Material Control and Accounting Requirements for Uranium Enrichment Facilities Authorized to Produce Special Nuclear Material of Low Strategic Significance" (1993). Topic Addressed / Discussed	Role / Purpose of Document in Relation to NUREG-1520
SER Appendix I, "Physical Protection."	Regulatory Guide 5.59 Part II, "Special Nuclear Material of Low Strategic Significance."	**Replacement of NUREG-1520.** NUREG-1520 does not apply to this part.

S1. General Information

Chapter 1 of the Safety Evaluation Report (SER) describes the NRC Staff's review of USEC's Application with respect to the facility and process description. *See* 10 C.F.R. §§ 30.33, 40.32, 70.22, 70.65; NUREG-1520. In regard to the general information in Chapter 1:

S1-1. Enrichment Process

A. The Environmental Impact Statement - Final Report (FEIS) states that USEC's Application is for a 3.5 million Separative Work Units (SWU) plant (FEIS at 1-2). The NRC Staff considered a 7 million SWU plant based on USEC's indication of a potential expansion to this capacity (FEIS at 1-3). What is the capacity of the ACP for the purposes of the Board's review of USEC's Application? Discuss any changes to either the SER or FEIS that need to be made to address a proposed expansion to 7 million SWU.

Staff Response: As noted in Section 1.1.3 of the SER, the Staff's review and safety and safeguards evaluation of the proposed plant is based on the nominal production capacity of 3.5 million SWU per year, which is based on the description of the plant provided by the applicant in its Application. However, as noted in Section 1.2 of the FEIS, the Applicant indicated in its ER the potential for future expansion to 7 million SWU per year if market conditions warrant. The Staff examined the potential environmental impacts of the plant based on the larger value of production capacity because, under the National Environmental Policy Act (NEPA), a federal agency should not only evaluate the environmental impacts of the proposed action, but also of any action that would be considered to be reasonably foreseeable.

Therefore, the Staff evaluated the environmental impacts of the larger production capacity as part of its current review.

The Applicant has not amended its Application to reflect an increase of production capacity from 3.5 million SWU to 3.8 million SWU, although it has made public announcements that such an increase is likely due to increased efficiency of its design. If the Applicant anticipates such a change for its license (if issued), it would evaluate the change through the change process identified in 10 C.F.R. § 70.72 and notify the Staff as appropriate. Any potential change to decommissioning funding would be addressed during the periodic review of the Decommissioning Funding Plan. There would be no need for an additional environmental review because the current evaluation in the FEIS is already bounding.

B. The NRC Staff proposes a license condition, which will require sixty (60) days notice before enrichment exceeds 5% U-235.

1. Explain the purpose behind this license condition.

Staff Response: The 60-day notification period provides the NRC Staff time to verify (through licensing review and/or inspection) that processes will be conducted safely and with sufficient safety margin at higher enrichments prior to production of higher-enrichment UF_6.

2. Indicate whether all safety and environmental analyses in the Staff's evaluation of USEC's Application have considered up to 10% enrichment.

Staff Response: Yes. The primary areas considered by the Staff due to increasing enrichment to 10% were criticality, material control and accounting, and production of tails for disposal.

3. Verify that no cylinder or containment of EU above 5% U-235 will be stored outside, show where and how EU > 5% will be stored, and indicate how this requirement will be reflected in the license.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

S1-2. Exemption Request

In regards to USEC's request for an exemption from the requirement to obtain liability insurance (SER at 1-13):

A. Discuss USEC's need for such an exemption, including the authority of DOE to exempt lessees from having liability insurance.

Staff Response: The NRC's regulations on financial qualifications for licensees and license applicants in 10 C.F.R. Part 140, Subpart B provide that:

> Each holder of a license issued under [10 C.F.R.] Parts 40 or 70 . . . for a uranium enrichment facility that involves the uses of source material or special nuclear material is required to have and maintain liability insurance [in] the type and in the amount the Commission considers appropriate to cover liability claims arising out of any occurrence within the United States that causes, within or outside the United States, bodily injury, sickness, disease, death, loss of or damage to property, or loss of use of property arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source material or special nuclear material. Proof of liability insurance must be filed with the Commission . . . before issuance of a license for a uranium enrichment facility under [10 C.F.R.] parts 40 or 70.

10 C.F.R. § 140.13b. The liability insurance "may be furnished and maintained in the form of . . . an effective policy of liability insurance from private sources," pursuant to 10 C.F.R. § 140.14(a)(1).

In the Application, the Applicant stated that, pursuant to an agreement between DOE and USEC, the ACP must be constructed on land leased by USEC from the DOE reservation at either the Portsmouth Gaseous Diffusion Plant (GDP) or the Paducah GDP. Application Rev.1 at 1-50. American Nuclear Insurers (ANI), the insurance pool that provides nuclear liability insurance, has declined to sell insurance to USEC for the operation of the ACP on an existing DOE site. Their unwillingness to sell insurance in this instance derives from the fact that the site on which the new USEC facility is being constructed is not a new "clean" site, but is instead the same site where previous DOE activities occurred. Letter, John Hoffman, American Nuclear Insurers to Donald J. Hatcher, USEC, Inc. (Oct. 10, 2006), submitted to the Board by Staff letter dated Jan. 29, 2007.

DOE has not "exempted" USEC from its obligation to obtain commercially-available liability insurance. Rather, the lease agreement between USEC and DOE provides that DOE will indemnify USEC against claims arising from nuclear incidents to the extent that USEC cannot obtain commercial insurance at reasonable rates. "Lease Agreement Between the United States Department of Energy and the United States Enrichment Corporation for the Gas Centrifuge Enrichment Plant," Appendix 1, December 1, 2006 ("Lease Agreement") at Section 10.1(d). The lease between DOE and USEC provides that USEC is

> indemnified under Section 170d of the Atomic Energy Act for liability claims arising out of any occurrence within the United States, causing, within or outside the United States, bodily injury, sickness, disease, or death, or loss of or damage to property, arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source or special nuclear material arising out of activities under the lease.

Id. DOE has confirmed that it will provide indemnification for the ACP. Letter from Larry W. Brown, Senior Advisor, Office of the Under Secretary, Department of Energy to Ron F. Green, Senior Vice President, USEC, Inc., March 7, 2005. Although DOE has not provided the Staff with an official legal position on its authority to indemnify USEC, as evident from its agreement to indemnify USEC, DOE has determined that it has the authority to do so pursuant to its authority under Section 170.d of the Atomic Energy Act of 1954. as Amended (AEA) to "enter into agreements of indemnification . . . with any person who may conduct activities under a contract with" DOE. 42 U.S.C. § 2210. Because DOE is the lead agency for administering its Price-Anderson obligations and commitments, the Staff has deferred to DOE's judgment in this area.

B. Provide the NRC Staff's interpretation of DOE's definition of "commercially available" liability insurance (SER at 1-13), the authority for this definition, and the historic precedent for such a reservation at other enrichment facilities.

Staff Response: NRC regulations require insurance from "private sources." 10 C.F.R. § 140.14(a)(1). The Staff considers private sources to mean non-government-funded or “commercially available.” The only currently operating enrichment facility is a DOE owned, contractor operated facility (the Paducah GDP). Under its lease with DOE, the contractor is indemnified by DOE under section 170d of the AEA. The recently licensed LES National Enrichment Facility (NEF) has obtained $1 million of standby liability insurance for construction of the facility. *See* 10 C.F.R. § 140.13. Prior to taking possession of source or special nuclear material, LES has committed to increasing its coverage to the maximum available. NEF Safety Analysis Report, Rev. 2 at 1.2-4 (July 2004) (ADAMS ML060680653). When issued, the LES license included the following condition related to liability insurance:

> The licensee shall provide proof of full liability insurance as required under 10 CFR 140.13b, at least 30 days prior to the planned date for obtaining licensed material. If the licensee is proposing to provide less than $300 million of liability insurance coverage, the licensee shall provide, to the NRC for review and approval, an evaluation supporting liability insurance coverage in amounts less than $300 million at least 120 days prior to the planned date for obtaining licensed material.

See Letter from Joseph G. Giitter, Chief, Special Projects Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards to Karl Gross, Licensing Manager, Louisiana Energy Services, “License for the Louisiana Energy Services National Enrichment Facility,” June 23, 2006 (ADAMS ML 061780384); *see also* NUREG-1827, “Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico,” page 1-11 (2005).

C. Discuss how DOE’s indemnification of USEC relates to claims against it and the United States Enrichment Corporation, and summarize what was done for liability insurance during the construction and operation of the lead cascade.

Staff Response: USEC, Inc. is the parent corporation of the United States Enrichment Corporation. Pursuant to the lease agreement, “the Corporation,” that is, “the United States Enrichment Corporation, its agents, representatives, and, if approved under the provisions of [the lease] its sublessees[9], successors, and assigns," is indemnified for "claims for public liability . . . which arise out of or in connection with activities under" the Lease. Lease Agreement at Sections 10.1(d)(1) and (2). This includes claims arising out of the construction, possession or operation of a production or utilization facility; transportation of source material, by-product material, or special nuclear material; possession, operation or use of a device utilizing special nuclear material or by-product material; or from activities related to nuclear waste. The lease covers activities for both the ACP and the Lead Cascade. Lease Agreement at 10.1. For certain claims, the Corporation agrees to waive certain defenses, including defenses as to the conduct of the claimant or the fault of the entities indemnified and any defense as to charitable or governmental immunity. Lease Agreement at Section 10.1(e).

As noted above, the current lease agreement between USEC and DOE covers Lead Cascade activities. However, the NRC's regulations do not require insurance coverage for test facilities. Therefore, the Staff neither requested nor received information regarding liability insurance coverage for the Lead Cascade.

S1-3. Additional Questions:

A. What is the likelihood of a tornado hitting the cylinder storage yards (SER § 1.3.3.3.2)?

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

B. How were the soils with 28% to 43% fines classified as a clay or silt? Clarify why the X-3346 Customer Service Building has a different Design Basis Earthquake than the rest of the ACP facility (SER at 1-30).

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

[9] USEC, Inc., the Applicant, is an approved sublessee. Lease Agreement at Section 14.3(d).

S2. Integrated Safety Analysis (ISA) and ISA Summary

Chapter 3 of the SER describes the NRC Staff's review of the ISA and ISA Summary. In regards to the ISA information contained in Chapter 3, please address the following:

S2-1. Sufficiency of Review Information

To help determine the sufficiency of the information in USEC's Application and the adequacy of the NRC Staff's review relating to the ISA, the Board directs the Staff to explain their evaluation of USEC's ISA as follows (providing examples for the ACP where relevant and appropriate):

A. Discuss the level of design details needed to assess USEC's ISA as documented in the internal memorandum and position statement of August 4, 2006, and Staff memoranda of September 13 and October 19, 2006[10]. This discussion[11] should highlight:

---

[10] The individuals who filed the September 13 and October 19 memoranda, along with some of their colleagues, have filed a formal differing professional opinion (DPO) (attached, but not publicly available) raising their concerns. Because a DPO has been filed, the differing viewpoints of the individual staff members are now being addressed through the NRC's formal DPO process. The DPO process is "a formal process for expressing differing professional opinions (DPOs) concerning issues directly related to the mission of NRC." Management Directive 10.159 at 1 (2004). The DPO process is intended to be an internal process. At the outset of the process, an Ad Hoc DPO Review Panel is appointed. Management Directive 10.159 Handbook at 6-8 (2004). After conducting a thorough review of the actions and opinions at issue, this panel will make a recommendation to the Office Director to whom the DPO has been assigned. *Id.* After receiving and reviewing the panel's recommendation, the Office Director will make a decision on the outcome of the DPO. *Id.* at 8. The Office Director's decision may be appealed to the Executive Director for Operations. The DPO process is separate from the hearing process.

[11] As the Board is aware, the position statement in the August 4 Memorandum represents the current policy position of management in NMSS. It was this policy position that provided the framework for the Staff's review of USEC's license application. Because the Board is charged with determining the adequacy of the Staff's review, the responses below focus primarily on the August 4 Memorandum. The Board has also requested information related to the September 13 and October 19 memoranda, which set forth the differing viewpoints of two members of the NRC staff. Although the differing views of the individual NRC staff members are best explained in the September 13 and October 19 memoranda and the DPO, the Staff has, in the responses to the Board's questions representing the current policy position, endeavored to fully respond to the Board's questions by summarizing all three memoranda, providing additional detail on the August 4 Memorandum setting out the current policy position, and presenting the prevailing Staff interpretation of the views in the September 13 and October 19 memoranda. In the Scheduling Order, the Board stated that "the parties may supplement their initial answers to these questions up until . . . March 5, 2007." Scheduling Order at 2. In the interest of presenting complete information to the Board, the Staff intends to supplement its response by affording the individuals who filed the DPO an opportunity to supplement the Staff's answer via affidavit by February 26, 2006 (the date by which USEC must supplement the Staff's response). Allowing the individuals to submit their viewpoints via affidavit is consistent with past Licensing Board and Appeal Board proceedings where the viewpoints of individual NRC staff members differed from the prevailing policy position. *See Louisiana Power & Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-803, 21 NRC 575, 580-81 (1985).

1. The applicable sections of 10 C.F.R. Part 70.

Staff Response: 10 C.F.R. Part 70, as well as 10 C.F.R. Part 40, is a performance-based regulation. For licensing a facility under 10 C.F.R. Part 70, technical information on the proposed equipment and facility must be provided in the application in accordance with 10 C.F.R. 70.22(a)(7), which states that each application shall contain:

> A description of equipment and facilities which will be used by the Applicant to protect health and minimize danger to life or property (such as handling devices, working areas, shields, measuring and monitoring instruments, devices for the disposal of radioactive effluents and wastes, storage facilities, criticality accident alarm systems, etc.).

The requirements for approval of an application are provided in 10 C.F.R. 70.23(a). These requirements state that an application will be approved upon a finding that the applicant is qualified, the proposed equipment and facilities are adequate to protect health and minimize danger to life or property, and the proposed procedures are adequate.

Additional requirements for approval of an application are provided in 10 C.F.R. § 70.66(a). These requirements state that an application for a license from an applicant subject to 10 C.F.R. § 70, Subpart H, will be approved if the Commission determines that the Applicant has complied with the requirements of 10 C.F.R. §§ 70.21, 70.22, 70.23, and 70.60 through 70.65.

In addition, 10 C.F.R. § 70.61 requires each applicant to evaluate, in an ISA performed in accordance with 10 C.F.R. § 70.62, compliance with the performance requirements in 10 C.F.R. § 70.61(b), 10 C.F.R. § 70.61(c), and 10 C.F.R. § 70.61(d). The regulations in 10 C.F.R. § 70.65 describe the requirements for the contents of the ISA summary that is required to be submitted with the application. The requirements in 10 C.F.R. § 70.65(b)(3) require that the ISA summary contain:

> A description of each process (defined as a single reasonably simple integrated unit operation within an overall production line) analyzed in the ISA in sufficient

> detail to understand the theory of operation; and, for each process, the hazards that were identified in the ISA pursuant to §70.62(c)(1)(i)-(iii) and a general description of the types of accident sequences.

The above requirement uses the term "single reasonably simple integrated unit operation within an overall production line." This language is meant to necessitate a sufficient discussion of the information needed to understand the safety implications of the system and the functions of system components used to ensure safety. A functional-level of design information is sufficient for review and to make the required such findings.

The regulations in 10 C.F.R. § 70.65(b)(6) require that the ISA summary contain:

> A list briefly describing each item relied on for safety which is identified pursuant to §70.61(e) in sufficient detail to understand their functions in relation to the performance requirements of §70.61.

Again, this regulatory citation specifies that a functional-level of design information is necessary and sufficient for review and to make the required findings.

Based on the information in the application and the ISA summary provided as required under 10 C.F.R. § 70.65, licensing decisions are made as required under 10 C.F.R. §§ 70.21, 70.22, 70.23, and 70.60 through 70.65. These decisions include a sufficiency determination with regard to compliance with the performance requirements, the baseline design criteria, defense-in-depth, and the adequacy of management measures.

In 10 C.F.R. Part 70 licensing pursuant to the current policy position as expressed in the August 4 Memorandum and consistent with historical NRC licensing practice, the Staff uses a reasonable assurance standard and focuses on the programmatic provisions of the Applicant's proposed activities. The term "programmatic" means a description of safety and administrative programs as well as structures, systems, and component designs at a functional programmatic level as opposed to a detailed, final design-level construction specifications level. This is reflected in the phraseology of the above-cited licensing requirements: "sufficient detail to understand the theory of operation," or a list "briefly describing each item relied on for safety ... in sufficient detail to understand their functions in relation to the performance requirements." "Programmatic" also refers to programmatic commitments to use specific codes and standards in the design of structures, systems, and components of the facility.

This approach is also reflected in the various chapters of NUREG-1520. Based on this understanding, the current policy position holds that licensing review needs to focus only on an applicant's programmatic and functional-level commitments. Consequently, the licensing decision is ultimately based on a sufficient level of detail to understand process system functions and how IROFS will provide reasonable assurance that they will perform their intended function and be reliable and effective.[12]

The reasonable assurance standard is applied to the programmatic, functional-level reviews such that the Staff decision pertains to a reasonable assurance that the ISA summary is complete and the licensee will follow its ISA approach and maintain it consistent with applicable regulations. The level of detail required for a licensing decision, pursuant to the reasonable assurance standard, therefore, does not require a finalized facility design or an

[12] The level of technical detail necessary for the Staff to assess the effectiveness and reliability of controls (including IROFS) depends on the degree to which the use of these controls is consistent with standard industry practice, the complexity of the controls as integrated into the process, the reliability required, and the degree to which the effectiveness and reliability are dependent on management measures such as inspection, testing and maintenance. Industry experience and the existence of relevant codes and standards help to simplify the Staff review.

Many of the controls to be used in the ACP facility have been used before in the GDPs and the Lead Cascade. Hence, their effectiveness and reliability are either known to the Staff or are presently being evaluated through operational experience. In addition, national consensus codes and standards may specify the design and operation of the control or components of the control in sufficient detail that a commitment to the code or standard will offer reasonable assurance that a required level of effectiveness and reliability is met.

Controls such as some passive controls or administrative controls (e.g., pressure vessels or combustible loading controls) do not require detailed descriptions. For more complex systems such as a detection and alarm system or a fire suppression system, the design basis of the system will allow the Staff to evaluate their effectiveness. The reliability of both types of systems will be assured through industry standards or national consensus.

absolute-level-of-assurance with a complete identification of all the details of IROFS and accident sequences. Instead only sufficient information has to be provided to understand the process and functions of items relied on for safety and to cultivate a reasonable assurance that the ISA summary is complete.

For uranium enrichment facilities, to ensure that an applicant's programs have been sufficiently implemented and commitments have been properly applied in the final facility design and in the constructed facility, the regulations in 10 C.F.R. § 40.41(g) and 10 C.F.R. §70.32(k) state that:

> No person may commence operation of a uranium enrichment facility until the Commission verifies through inspection that the facility has been constructed in accordance with the requirements of the license.

This requirement, applied through operational readiness review inspections rather than through licensing reviews, will ensure that the programmatic and functional-level commitments made by the licensee, as well as any license conditions imposed by the Board, are properly applied in an as-built facility. Inspections, therefore, are intended to focus on the final design of the facility and the procedures that have been prepared to implement the licensee's commitments that are reflected in the license.

In the development of the performance requirements in 10 C.F.R. Part 70, it was anticipated that, in the future, changes will likely be made to corresponding operational facility design and processes. A methodology for addressing these changes is described in 10 C.F.R. § 70.72. For a uranium enrichment facility, the licensee may make changes to its design after receiving its license during the construction phase or after operations begin under the provisions of 10 C.F.R.§ 70.72.

2. Differences in the two positions established by the referenced memos, emphasizing the design requirements for hazard identification and description, accident sequence, IROFS, and management needed to meet the reasonable assurance standard.

Staff Response: There are several areas in the September 13 and October 19 memoranda that differ from the guidance in the Robert C. Pierson memorandum of August 4, 2006. These areas involve the following: (a) Use of the term "programmatic;" (b) Use of the term "complete;" (c) Use of information obtained from visits to Urenco's operational uranium enrichment facility in Almelo, The Netherlands; (d) uncited regulations; and (e) the previous Atomic Safety and Licensing Board review of the LES application.

In the August 4 Memorandum, which represents the current Staff policy position, the term "programmatic" refers to a description of safety and administrative programs as well as structure, system, and component designs at a functional, programmatic level, as opposed to detailed, final design-level construction specifications. "Programmatic" also refers to program commitments to use specific codes and standards in the design of structures, systems, and components of the facility. This is reflected in the above-referenced licensing requirements that talk about, "sufficient detail to understand the theory of operation," or a list "briefly describing each item relied on for safety ... in sufficient detail to understand their functions in relation to the performance requirements." The use of "programmatic" in terms of a functional-level of detail is consistent with the regulatory requirements.

In the September 13 and October 19 memoranda, the authors limited their definition of "programmatic commitments" to those commitments pertaining to an applicant's safety programs. The Staff interprets this definition to be more general than the functional-level of detail required by the regulations.

In the August 4 Memorandum, the reasonable assurance standard is discussed in the context of the completeness of the ISA summary. The memorandum states, as follows:

> The reasonable assurance standard is applied such that the Staff decision pertains to a reasonable assurance that the integrated safety analysis summary is complete and the licensee will follow its integrated safety analysis approach and maintain it consistent with the regulations. The level of detail required for a licensing decision, therefore, does not require a final facility design or an absolutely complete identification of all items relied on for safety and accident sequences, but instead sufficient information has to be provided to understand the process and functions of items relied on for safety and reasonable assurance that the ISA summary is complete.

Pursuant to the current policy position, the term “complete” should be used in the context of reasonable assurance, not in terms of an “absolute-level” of completeness evidenced by complete and final design detail.

The authors of the September 13, 2006, and October 19, 2006, memoranda, however, disagree with the prevailing policy position as to what level of design information would be a necessary prerequisite for a finding of a reasonable assurance that the ISA is complete.

As reflected in the Staff policy position outlined in the August 4 Memorandum, the Staff’s review of an applicant’s ISA does not involve a 100 percent review, but includes samples of material prepared in a vertical slice fashion. This vertical slice review concept is consistent with the review approach presented in NUREG-1520 and NUREG-1513.

The August 4 memorandum states, “The fact that the LES plant is based on a facility currently operating in Europe has no effect on the licensing basis used to issue the LES license. The licensing bases for a uranium enrichment facility are solely the regulations in 10 C.F.R. Parts 40 and 70 and are not dependent on operations of similar facilities located elsewhere.” The authors of the September 13 and October 19 memoranda, however, appear to believe that this statement means that Staff received no benefits from Urenco facility visits, discussions with Urenco Staff, or reviews of detailed information supporting the ISA in Europe. This is not the case. The visits to Urenco facilities and the discussions with Urenco Staff enhanced the reviews of the Application and increased the understanding of the potential risks of the facility.

However, these visits did not affect the licensing bases required to make the findings needed to issue the license. The licensing bases and approach are dictated by the requirements in 10 C.F.R. Parts 40 and 70, regardless of whether a licensing action involves an entirely new facility or a similar facility operating elsewhere.

The authors of the September 13 and October 19 memoranda also discuss other regulatory citations that they believe should have been referenced in the August 4 memorandum. While the Staff acknowledges that there are other regulations in 10 C.F.R. Part 70 that are not cited in the August 4 Memorandum that apply to licensing reviews under Part 70, the prevailing Staff viewpoint is that these other regulations do not directly pertain to the specific issue of the required level of detail needed in performing a licensing review.

The August 4 Memorandum references the Atomic Safety and Licensing Board's decisions in the LES case and states that these decisions validate the approach taken by the Staff. The authors of the September 13 and October 19 memoranda infer that the fact that the Board made a positive finding in the LES case is inconclusive because the Board decisions did not specifically address the design issue. The statements in regard to the Board decisions in the LES proceeding in the August 4 Memorandum are based on the fact that the LES Board evaluated the Staff's review and had access to the entire application submitted by LES, as well as the testimony provided in the contested and uncontested portions of the licensing hearing. The Staff also provided an extensive discussion on the review approaches described in NUREG-1520. In addressing the adequacy of the Staff's technical and environmental review, the Board never questioned the adequacy of the LES submittal. Without a sufficient application, the Staff could never have performed an adequate technical and environmental review.
The prevailing Staff opinion, therefore, is that the LES Board did accept the Staff's approach to reviewing the LES application even though the level of detail was not expressly addressed.

3. The degree to which the position expressed in the August 4, 2006 memorandum would require a rule change.

Staff Response: As discussed in the responses to Items 1 and 2 above, the position expressed in the August 4 Memorandum, is entirely consistent with the regulations in 10 C.F.R. Part 70 and no rule changes are necessary. The authors of the September 13 and October 19 memoranda disagree with this position.

4. The relationship between the Staff's position on the required level of design detail and the guidance provided in NUREG-1520.

Staff Response: Discussions in NUREG-1520 related to the ISA appear to the majority of the Staff to be primarily focused on the existing operating fuel cycle facilities, where there are detailed, final, as-constructed designs. However, for new facilities, especially uranium enrichment plants that cannot begin construction until a license is issued, it is unreasonable to assume that final, to-be-constructed designs will be available at the time an Application is submitted. Furthermore, only the functional-level information is required by the regulations to ensure a sufficient basis for review. Based on the lessons learned from the LES review, in future revisions to the SRP, the Staff plans to provide clarifying guidance on the necessary level of detail for new facilities. The authors of the September 13 and October 19 memoranda would not propose to change the SRP guidance.

5. The rationale for not requiring design details beyond programmatic commitments for this license review.

Staff Response: Pursuant to the policy position expressed in the August 4 Memorandum, the rationale for not requiring design details beyond programmatic, functional-level commitments is based on the regulatory requirements discussed in the above response to Question S2-1.A.1 above. The regulations require a functional-level of design detail for structures, systems, and components with appropriate programmatic commitments to use specific codes and standards. The Staff also considers that an ISA summary should be considered complete as per the reasonable assurance standard, instead of a more rigorous level of assurance.

The approach of using a reasonable assurance standard and programmatic functional-level commitments to make the licensing decision for a uranium enrichment facility was used in both the USEC Inc. Lead Cascade review and the LES review. In the uncontested portion of the LES licensing proceeding, the Board concluded that based on its review, the Staff had adequately performed its environmental and safety reviews and a license could be issued. The majority of the Staff believes that the Board decisions confirm that the above approach meets the legal requirements in 10 C.F.R. Parts 40 and 70.

6. Options to assure that final design details and/or design changes are consistent with the programmatic commitments, and that the IROFS have been incorporated into the ACP facility.

Staff Response: Under the regulations in 10 C.F.R. § 40.41(g) and 10 C.F.R. § 70.32(k), the NRC must verify through inspection that the facility has been constructed in accordance with the requirements of the license. (Note that for fuel cycle facilities, license application documents, except the ISA summary, are specifically incorporated as license commitments. Under 10 C.F.R. § 70.65(b), the ISA summary does not become part of the license.) The Staff or Licensing Board, or both, may also impose additional licensing conditions deemed appropriate to ensure that the facility will be built and operated safely. If a license to construct and operate the facility is granted, the Staff will conduct an Operational Readiness Review (ORR) to confirm that the applicant (now licensee) has built the facility in accordance with the application and any incorporated license conditions. Only then will the licensee be allowed to operate the facility. It is through this inspection process that the Staff will ensure that the design and construction of the facility meet the licensee's commitments in its application. At that time, the Staff will also ensure that the licensee is ready to begin operations and has the appropriate procedures in place to operate the plant safely and in accordance with all applicable NRC requirements. This review also ensures that IROFS have been incorporated in the facility as specified in the license application. These inspections would be conducted by Region II inspection staff with assistance from Headquarters technical staff as necessary.

In the development of the performance requirements in 10 C.F.R. Part 70, it was anticipated that, in the future, changes will be made to the facility design and processes. Therefore, a process for addressing these changes is described in 10 C.F.R. § 70.72. For a uranium enrichment facility, the licensee may make changes to its design after receiving its license during the construction phase and after operations begin. These changes, therefore, need to be submitted, as necessary, and reviewed in accordance with 10 C.F.R.§ 70.72. The annual update submittals required under 10 C.F.R. § 70.72 would be reviewed by the Staff.

The policy position reflected in the August 4 Memorandum considers that ORR inspections and the 10 C.F.R.§ 70.72 change process provide adequate assurance that the facility will be designed, constructed, and operated in accordance with the license commitments and all applicable regulatory requirements.

B. Verify and explain why the most critical criticality accident is associated with product withdrawal and cylinder storage of EUF_6 at 10% enrichment.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

C. Describe the step-by-step process in the development of ISA with respect to the flow chart shown on SER Figure 3-1 (SER at 3-9 to 3-12). Illustrate the ISA process by following through the steps using the most critical of the accident sequences listed in the SER (<u>see</u> SER at A-18) (<u>i.e.</u>, criticality in EUF_6 cylinders during storage). Illustrate the IROFS in the accident sequence and controls relied on to meet the double contingency principle.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

D. Explain how the ISA Summary was prepared.

Staff Response: The ISA Summary was prepared in accordance with the requirements of 10 C.F.R. § 70.65(b)(1-9). The ISA Summary was prepared by an ISA Team which is described in Chapter 4 of the ISA Summary, "Integrated Safety Analysis Methods". The ISA team is composed of a number of individuals that serve as core members and subject matter experts in the design, operations, criticality, and accident areas of the ACP. The team leaders also have experience and expertise in hazard and accident analysis. The team performed the ISA using the preliminary hazard analysis and the (WI/CL) method for hazard identification and hazard evaluation and used the results of these analyses to prepare the ISA Summary.

The presentation of the accident evaluation followed the example in Table A-7 of the SRP (NUREG-1520) and each accident sequence included an accident identifier, causes, the unprevented likelihood, unmitigated consequence levels, preventive controls (including IROFS and defense-in-depth controls), mitigative IROFS (including IROFS and defense-in-depth controls), the prevented likelihood, the mitigated consequence level, and the risk classification. Estimations of numerical event probability, the probability of failure on demand of IROFS, consequence reduction from mitigative IROFS, and numerical values of consequences were provided in other appendices. In addition, the Applicant also demonstrated compliance with 10 C.F.R. § 70.61(c) regarding environmental consequences and provided a separate addendum for Interim Production.

The rest of the ISA Summary contained sections of text that were required by 10 C.F.R. § 70.64(b) and are consistent with the guidance in the SRP.

The ISA Summary was prepared in accordance with the guidance contained in Chapter 3.0 of NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, and NUREG-1513, Integrated Safety Analysis Guidance Document.

The information within the ISA was arranged into the following major sections:

- Chapter 1, Site Description
- Chapter 2, Plant Description
- Chapter 3, Process Description
- Chapter 4, Integrated Safety Analysis
- Chapter 5, Hazardous Material Inventory and Location
- Chapter 6, Accident Sequences
- Chapter 7, Selected Controls
- Appendix A, Methodology Tables
- Appendix B, Hazard Identification Tables
- Appendix C, Hazard Evaluation Tables
- Appendix D, Consequence Tables
- Appendix E, Event Frequency Development
- Appendix F, Risk Reduction - Preventive and Mitigative Values of IROFS
- Appendix G, Classified Submittal
- Appendix H, (Reserved for future use)
- Appendix I, Environmental Consequences
- Appendix K, Evaluation of Mitigative IROFS Failure

The ISA Summary was reviewed to ensure that the content was sufficient to meet the requirements of 10 C.F.R. § 70.65(b).

E. Demonstrate how the hazard evaluation methods (i.e. Preliminary Hazard Analysis and What if/Checklist) were derived from the use of Figure A.1 of NUREG-1513, Integrated Safety Analysis Guidance Document (see SER § 3.3.2.1). Discuss whether decommissioning was considered as an activity in the hazard identification and evaluation process, and, if so, show how it was incorporated, and, if not, explain the reasons for not including it in the evaluation.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

F. Explain how USEC's ISA commitments will be incorporated into its license and how the commitments will be tracked during construction, operation, and decommissioning.

Staff Response: The standard practice for licensing fuel cycle facilities is to directly incorporate license application documents into the license by tie-down references. In this way, all the Applicant's commitments documented in the licensing documents become enforceable. License conditions may not be changed without a license amendment. Other licensee commitments are binding subject to the licensee's ability to make changes pursuant to the 10 C.F.R. § 70.72 change process. Since the ISA summary, under 10 C.F.R.§ § 70.65(b), is not incorporated into the license, Staff ensured in the licensing review that commitments related to IROFS were documented in the Application (see Application Section 3.1), documented in the tie-down references in the license.

As described in the response to Question S2-1. A.6., the NRC will inspect the ACP during the construction and operations phases of the ACP facility to ensure the licensee has adequately implemented programs, processes and procedures to design, construct, install, test and operate the safety significant structures, equipment and components in a manner that protects the health and safety of workers, the public and the environment as required by NRC regulations and the license. This will be accomplished as described by the referenced IMCs.

The licensee is required to conduct and maintain an ISA by 10 C.F.R. § 70.62(c). The NRC's construction and operations inspection program includes a high level review and assessment of the licensee's conduct and maintenance (e.g., commitments) of the ISA and an in-depth review and assessment of selected elements of the ISA (e.g., IROFS and related management measures) based on safety/risk significance, past performance, significant changes and other safety related characteristics that may distinguish more significant elements from others. These in-depth reviews and assessments usually focus on criticality safety, radiation safety, fire safety, and chemical safety aspects of operations.

Regional staff will develop inspection plans to address specific license commitments. The region will track the findings and conclusions from these inspections as a part of the process to verify that the licensee has adequately met the specified license commitments. Inspections will be scheduled and appropriate plans prepared based upon the licensee's construction schedule; a prioritization of those safety-related structural features, equipment, and components that should be inspected; and available resources. The inspection plans will be modified as necessary to reflect emergent issues that surface during the inspection process so that proper inspection attention can be focused on critical areas.

A similar inspection approach will be used during the decommissioning phase of the facility.

S2-2. Implementation of Baseline Design Criteria (BDC), and Defense-in-Depth

10 C.F.R. § 70.64(a) requires each new facility or new process to address ten baseline design criterion. In regards to the NRC Staff's review of USEC's submittal:

A. Clarify, using one or more processes as an example, how each of the BDC were addressed in the design of the selected process.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

B. Discuss the relationship between defense-in-depth and IROFS. Do this by showing several component design example items that are considered defense-in-depth but not IROFS; ones considered IROFS but not defense-in-depth; and ones that are considered both. Clarify how adherence to the double contingency principle relates to both of these two controls.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

S3. Radiation Protection (RP)

Chapter 4 of the SER describes the NRC Staff's review of information relating to USEC's RP program. In regards to the Staff's review of USEC's RP program, please address the following:

A. Discuss how USEC's ISA commitments to an As Low as Reasonably Achievable (ALARA) program will be incorporated into its license and how the Staff plans to manage and track implementation of these commitments during construction, operation, and decommissioning.

Staff Response: The Staff does not need to separately incorporate USEC's ISA commitments to an ALARA program into the its license, because it is already required by 10 C.F.R.§ 20.1101(b), which provides that "the licensee shall use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonable achievable (ALARA)." USEC provided a description of its ALARA program in Section 4.2 of the Application (USEC, 2006), and in Section 4.3.2 of the SER the Staff found the ALARA program to be consistent with the guidance in Chapter 4 of NUREG-1520. However, the standard practice for licensing fuel cycle facilities is to directly incorporate license application documents into the license by tie-down references. In this way, all the applicant's commitments documented in the licensing documents, including those related to the ALARA program, become enforceable. License conditions may not be changed without a license amendment. Other licensee commitments are binding subject to the licensee's ability to make changes pursuant to the 10 C.F.R. § 70.72 change process. Since the ISA Summary, under 10 C.F.R. § 70.65(b), is not incorporated into the license, Staff ensured in the licensing review that commitments related to IROFS were documented in the Application (see Application Section 3.1), which would be documented in the tie-down conditions in the license. Commitments related to ALARA are documented in Section 4.2 of the Application.

The inspection process and procedures described in the response to Question S2-1.A.6. include a review and assessment of the licensee's radiation protection program. The licensee's ALARA program is reviewed and assessed as a part of this effort.

Regional staff will develop inspection plans to address specific license commitments. The region will track the findings and conclusions from these inspections as a part of the process to verify that the licensee has adequately met the specified license commitments. Inspections will be scheduled and appropriate plans prepared based upon the licensee's construction schedule, a prioritization of the safety-related structural features, equipment, and components that should be inspected, and available resources. The inspection plans will be modified as necessary to reflect emergent issues that surface during the inspection process so that proper inspection attention will remain focused on the most critical areas.

A similar inspection approach will be used during the decommissioning phase of the facility.

B. USEC has requested two exemptions pursuant to 10 C.F.R. Part 20 relating to its RP program: (1) exemption from posting and labeling each container of licensed material required by 10 C.F.R. § 20.1904; and (2) exemption from the requirements of 10 C.F.R. § 20.1601(a) that each high radiation area be controlled with devices to reduce exposures. Please address the following:

1. Labeling Exemptions (SER at 1-11, 4-15):

(a) Expand upon the discussion of the two labeling exemptions requested (i.e., one for labeling cylinders being transported inside the Portsmouth Gaseous Diffusion Plant (PORTS) boundary and one for labeling containers located in the Restricted Areas within the ACP). In this discussion, highlight:

(i) types and numbers of containers that will be in use during normal operations;
(ii) locations of the Restricted Areas and PORTS boundary;
(iii) frequency of movement and number of cylinders to be moved;
(iv) the definition of when a cylinder is "attended";
(v) why labeling the large number of cylinders would desensitize workers.

(b) Detail the survey to be performed when containers are removed from contaminated or potentially contaminated areas and how this survey will prevent the spread of contamination.

(c) Summarize any other considerations by the Staff in finding that USEC's request meets the exemption standards in 10 C.F.R. §§ 40.14 and 70.17.

Staff Response: (a) While the Applicant has requested two exemptions to the labeling requirements under 10 C.F.R.§ 20.1904 in fact only one exemption is needed, that being the exemption from labeling containers located in the Restricted Areas within the ACP. As explained by the Staff in SER Section 4.3.8, 10 C.F.R. § 20.1905(c) already exempts containers from 10 C.F.R. § 20.1904, if the containers are attended by an individual who takes the precautions necessary to prevent the exposure of individuals in excess of the established limits. The Staff will inspect the Applicant's procedures for moving and attending to containers that have not been labeled in accordance with 10 C.F.R. § 20.1904 as part of the ORR.

The locations of the restricted areas and PORTS boundary can be seen in Figure 1-1 of the Emergency Plan which shows the U.S. Department of Energy reservation.

10 C.F.R.§ 20.1904 requires that each container of licensed material bear a durable, clearly visible label such that the radionuclide(s) present, the quantity of radioactivity, radiation levels, kinds of materials, mass, and enrichment be identified. For the labeling exemption USEC has proposed to post one sign stating that "every container in this area may contain radioactive material", rather than labeling each container. Also, USEC has committed by procedure to perform a survey when containers are to be removed from contaminated or potentially contaminated areas so as to ensure that contamination is not spread around the reservation. The Staff will inspect its procedure for removing containers from restricted areas as part of its ORR.

(a)(i) Based on experience at other operating fuel cycle facilities, the Staff expects there to be multiple containers used ranging in size from very large to small. The number will range from several hundred to thousands depending on the operating status of the facility. In addition to UF_6 cylinders (48X, 30B, etc.) discussed in response to question (iii), UF_6 samples will be collected and transported for analysis and traps used to collect residual UF_6 and its reaction products. Based on GDP operational experience, it is estimated between 50 and 60 samples will be collected on a daily basis. Samples include vent samples, cascade assay measurements, and confirmatory measurements of tails, product, and feed cylinders. The actual number of samples will vary based on operational needs. This does not include various radiation protection, waste management, or environmental samples collected to demonstrate compliance with other ACP Programs.

In addition, satellite accumulation areas will have containers containing contaminated wastes generated during operation and maintenance activities. Prior to removal from areas controlled for removable contamination, containers will be surveyed, transported by a trained individual, and stored in approved locations.

(a)(ii) USEC provided a detailed description of its controlled area at the ACP site in Section 2.41, page 2-47, of its ISA Summary. A detailed description of its restricted areas can be found in Section 2.42, pages 2-48 thru 2-56, of its ISA Summary. ACP restricted areas are located within the PORTS reservation boundary. With the exception of two cylinder yards that are separately fenced to prevent unauthorized access, these areas are also within the ACP controlled access area. The locations of the restricted areas and PORTS boundary can be seen in Figure 1-1 of the EP which shows the DOE reservation.

(a)(iii) Cylinders will be transported on a daily basis depending on operational needs. At a minimum, this will include approximately:

- Tails cylinders to the tails storage yard (13/week);
- Feed cylinders moved to the Feed Oven (19/week);
- Emptied (heeled) cylinders to decay in storage (19/week);
- Filled Product cylinders moved from withdrawal stations (3/week);
- Product cylinder transferred to Customer Service facility for transfer to customer cylinders (3/week);
- Empty customer cylinders arriving (20/week); and
- Customer cylinders being shipped (20/week).

(a)(iv) 10 C.F.R. Part 20 does not define the term “attended” so the Staff uses the common understanding of the term. Accordingly, the Staff would expect that an individual(s) would be physically present in the area of the container or able to control access to the container. Such personnel shall be trained to the appropriate level of Radiological worker training outlined in Chapters 4.5 and 11.3 of the ACP Application. This issue will be addressed by the Applicant’s detailed radiation protection procedures that will be reviewed during the ORR.

(a)(v) The large number of potential containers that would be required to be labeled in any area would confront a worker with a sea of radiation signs. These signs as required by 10 C.F.R. § 20.1901, would use the colors magenta, or purple, or black on yellow background.

These colors were chosen by the NRC because they are very bright and eye catching. There is a concern that wherever a worker looked they would see radiation labels with these bright colors everywhere, such that they would cease to have any safety meaning. As such, workers would be desensitized to radiation signs. This exemption is essentially identical to exemptions previously approved for the gaseous diffusion plants.

(b) The Applicant's procedure for performing surveys when containers are removed from contaminated or potentially contaminated areas has not been developed. These surveys are commonplace at NRC licensed facilities and the Applicant has considerable experience with such surveys at the GDPs. The detailed radiation protection procedures including those for performing surveys will be inspected during the ORR. However, as stated in NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation," in Part 20, the meaning of "Survey" is more than the measurement of dose rates using a survey instrument. In Part 20, the meaning of survey is broader and includes any activity using available relevant information, including data obtained form field measurements, to assess the radiation hazards. As noted in NUREG-1520, performing a survey in the field with a survey instrument without assessing the resulting data to evaluate hazards would not be considered as having satisfied the requirement to perform an adequate survey. Normally a survey to move a container would include monitoring the dose rate from the container with a radiation detection instrument and a wipe survey to determine if the container had loose contamination that needed remediation. Action levels would be specified in the radiation protection procedure to allow workers to assess the results. If the level monitored is below the action levels, workers would be allowed to move the container. If not, the container would not be allowed to be moved. Such surveys, as defined in Part 20, are sufficient to prevent the spread of contamination because of the present hazard evaluation implementations.

(c) In evaluating the exemption request, the Staff did not consider any issues other than those discussed above or in SER Section 4.3.8. As noted in the SER, experience at facilities that have received the exemption from the labeling requirement demonstrates that the Applicant's request will provide an equivalent amount of safety and will not result in an undue hazard to individuals.

2. Alternative Methods for Controlling Access to High Radiation Areas (HRA) (SER at 1-12, 4-15):
(a) Elaborate on USEC's need to request approval for alternative methods for controlling access to high radiation areas, including the burden with meeting the current regulation, the advantage of USEC's proposed method, and the details of the physical and administrative controls proposed by USEC to prevent inadvertent or unauthorized access to HRAs and very HRAs.
(b) The Staff's justification for approving alternatives to 10 C.F.R. § 20.1601(a) lies, in part, on the fact that HRA are controlled by Radiation Work Permits (RWP). Explain how the RWP procedure works with selected examples, and comment on those cases, if any, that the requirement for a RWP could be waived, what procedures would need to be followed to activate a wavier of these requirements, and who could authorize such a waiver and under what circumstances.

Staff Response: (a) Since the alternative methods of controlling access to High Radiation Areas ("HRAs") is not an exemption request, USEC is not required to explain the need or explain the burden of meeting 10 C.F.R.§ 20.1601(a). Under 10 C.F.R. § 20.1601(c), a licensee may propose alternative methods for controlling access to HRAs. These methods do not represent an exemption from 10 C.F.R. § 20.1601, but an alternative means for complying with it. To meet the requirements of 10 C.F.R. § 20.1601(c), USEC only needs to propose an effective method of controlling access to HRAs. In order to accept the alternative methods, the Staff need only make a finding that the alternative method provides the worker with sufficient information about radiation levels in the area and prevents unauthorized or inadvertent access. As explained in Section 4.3 of the SER, the Applicant has proposed that each HRA will be conspicuously posted, "Caution, High Radiation Area," and entrance into each area will be controlled by a radiation work permit (RWP). Also, for areas with radiation levels exceeding 100 mrem/hr at 30 cm but less than 1 rem/hour at 30 cm, USEC will use physical and/or administrative controls similar to those outlined in NRC Regulatory Guide 8.38, Section C.2.4, to prevent inadvertent or unauthorized access to HRAs, including the following:

1. Radiological worker II training;
2. Worker's signature on the RWP;
3. Personnel TLDs and supplemental dosimeters;
4. Survey meter or dose rate indicating device available at the work area; and
5. Periodic radiation surveys performed by Radiation Protection personnel in accordance with RWPs.

ACP HRAs with dose rates greater then 1 rem/hour will be locked and the key is controlled to prevent inadvertent or unauthorized access, consistent with the requirements of 10 C.F.R. § 20.1601. Very High Radiation Areas are not anticipated at the ACP.

RWPs are used by the majority of the fuel cycle and reactor licensees. The Applicant has considerable experience using RWPs at its GDPs. RWPs are typically used for activities such as entry into radiation and contamination areas, temporary jobs, and non routine work activities. Also the Applicant will implement physical and administrative controls to prevent inadvertent or unauthorized access to HRAs and Very High Radiation Areas. Thus, the information provided to the worker by the HRA area posting and RWP system provides information about the radiation levels, the RWP also provides information on safety equipment needed to enter the HRA, and the implementation of physical and administrative controls by USEC will prevent inadvertent or unauthorized access to HRAs and Very High Radiation Areas.

Therefore, the Staff found in Section 4.3 of its SER that the Applicant had met the requirements of 10 C.F.R. § 20.1601(c).

(b) As explained by USEC in Section 4.4.2 of the Application, RWPs are a basic implementing tool by which radiological controls are established. The purpose of an RWP is to provide information to the worker concerning protective clothing, job/task identification, and special instructions, such as radiological hold points. Radiological surveys that supplement RWPs provide information regarding radiation and contamination levels. Qualified health physics (HP) personnel are authorized to approve, issue, update, revise, and close RWPs. An RWP at the ACP may be issued for any period up to one year, based on the stability and predictability of changes in the radiological conditions of the work area. RWPs are normally closed upon job completion. HP management reviews the RWP closure package to ensure appropriate actions have been taken.

In terms of waivers to its RWP process, USEC has proposed that continuous HP coverage may be used in lieu of RWPs when approved by the Radiation Protection Manager. Qualified technicians are authorized to provide continuous radiological coverage in lieu of an RWP for short duration (less than one shift), non-complex tasks. When continuous HP coverage is used, requirements normally specified on an RWP are communicated to the worker verbally, and a trained HP monitor is physically present during this work. The RPM may exempt the requirement for an RWP as specified in approved procedures.

S4. <u>Nuclear Criticality Safety (NCS)</u>

Chapter 5 of the SER describes the NRC Staff's review of USEC's NCS program. Please address the following:

A. In regards to criticality monitoring when enriching up to 10% U-235, discuss, in detail:

1. <u>Enrichment processes</u>:

(a) Potential process steps and areas where criticality could become an issue under normal and accident conditions as identified by the Nuclear Criticality Safety Evaluation (NCSE).

(b) Clarify how many stages of centrifuges will make up a cascade; how many cascades will be installed in each process building and whether they will be placed in one large area or whether the building will be divided into rooms for various centrifuge cascades; and, what is the potential for criticality at any given portion of the operational process.

(c) Clarify that all criticality analyses and monitoring evaluations were performed assuming 10% enrichment, or, for each situation that they were not, demonstrate that lower enrichment levels are equal to or more critical than 10% enrichment.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

2. Cylinder storage and the Criticality Accidents Alarm System (CAAS) exemption (SER § 5.3.6):
(a) The difference between the exceptions derived from NCSE with the exemptions requested by USEC for only one of the other areas designated by NCSE (i.e. UF_6 cylinder yards).
(b) Demonstrate that the exemption criteria in 10 C.F.R. § 70.17 is applicable for the UF_6 cylinder yards and discuss how the CAAS program would assure that any near criticality could be detected and adverted at the other identified areas.
(c) Explain how cylinders with depleted uranium (DU) will be stored on site, giving the approximate maximum storage capacity at each location, length of anticipated storage time at each location, and ultimate disposition of the DU.
(d) Show where cylinders containing uranium enriched to 5 percent or less weight U-235 will be stored on site, giving the approximate maximum storage capacity at each location, length of anticipated storage time at each location, and ultimate disposition of the enriched product.
(e) Show where cylinders containing uranium enriched between 5 and 10 percent weight U-235 will be stored on site, giving the approximate maximum storage capacity at each location, length of anticipated storage time at each location, and ultimate disposition of the enriched product.
(f) In regards to the CAAS exemption:
    (i) Describe in greater detail how the integrity of the cylinders and vehicle handling practices make the occurrence of a large cylinder breach very unlikely, and demonstrate how the response time will ensure that the cylinders will not accumulate sufficient moderator to make criticality possible with 10% enrichment (SER at 5-27);
    (ii) Demonstrate in greater detail how the Staff's independent analysis yields the results shown herein (SER at 5-28);
    (iii) Explain how USEC's commitment to store cylinders with more than 5% enrichment indoors and with CAAS coverage is to be documented in the license (SER at 5-29).

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

B. In addition, please address the following:

1. The first paragraph of SER page 5-6 states that walk-throughs will be performed annually, while the second paragraph discusses monthly walk-throughs. Conversely, the NUREG-1520 acceptance criteria states that walk-throughs should be performed at least every two weeks. Which

of the above is the proposed schedule for the ACP? Clarify the Staff's justification for accepting USEC's reduced walk-through schedule.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

2. Clarify why the acceptance criteria in NUREG-1520, Section 5.4.3.4.6(3) is unnecessary, and why there is no need for the specific requirements to use certain methodologies and practices as discussed in the acceptance criteria (SER at 5-30).

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

S5. Chemical Process Safety

Chapter 6 of the SER describes the NRC Staff's review of information in USEC's Application related to chemical process safety. Please address the following:

A. Part of the Staff's responsibility is to oversee chemical safety issues of licensed materials related to the risks to workers, public, and the environment; In this regard, summarize the chemicals that will likely be used or formed during the enrichment process at the ACP and which chemicals are related to the storage, handling, and processing of licensed materials.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

B. Compare potential ACP-related chemicals to those chemical impacts currently detected in the soils, surface water, sediments, and groundwater in the areas downgradient of the ACP processes at the site so as to ascertain if the chemical safety related evaluations for ACP will be successful in discriminating past actions from plant performance, including unmitigated accident sequences.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

C. Discuss USEC's Corrective Action Program (SER at 6-13), and how it will address any unacceptable performance deficiency discussed as part of Staff's review in NUREG-1520 (NUREG-1520 at 6-3).

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

D. Verify that the reference to Appendix F on SER page 6-15 relates to USEC's ISA Summary and not Appendix F of the SER. Summarize the unmitigated risk of sequences that could exceed the performance sequence of 10 C.F.R. Part 70, Subpart H, and discuss the Staff's review of the selected consequence accident scenarios, focusing on its confirmation that the chemical events that could exceed the performance requirements of 10 C.F.R. Part 70 were addressed.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

S6. <u>Fire Safety</u>

Chapter 7 of the SER describes the NRC Staff's review of information in the application related to fire safety.

A. The Staff states that USEC will use most of the applicable codes presented in Table 7-1 will be used by USEC "except as justified through documentation" (SER at 7-2 to 7-4). What is the criteria for authorizing any departure from the established code, who reviews this justification, and who has the responsibility and authority to approve any such justification?

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

B. Provide some examples of cases where the code standard would not be used, the documentation that would be provided to support the deviation from the standard, and the approval process that would be followed in such cases.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

S7. <u>Environmental Monitoring</u>

Chapter 9 of the SER documents the NRC Staff's review of USEC's environmental monitoring protection plan. To help the Board evaluate the thoroughness of the Staff's review, please address the following:

A. Chapter 9 of the SER states that the acceptance criteria relating to the Staff's review of USEC's Environmental Report (ER) is not applicable since the Staff has prepared an FEIS (SER at 9-2). However, the review criteria in NUREG-1520, Chapter 9, states that the Staff will determine whether an applicant has submitted an ER that is adequate to prepare either an Environmental Assessment/Finding of No Significant Impact or an EIS. Given that the FEIS submitted by the Staff is required to be its independent assessment of the environmental impact, discuss the adequacy of USEC's ER in relationship to the acceptance criteria in NUREG-1520 § 9.4.3.1.1, or reference where an assessment of the adequacy has been done in the SER or in the FEIS.

Staff Response: The criteria for an ER provided in Section 9.4.3.1.1 refers the reviewer to the NRC environmental regulations found in 10 C.F.R. Part 51, specifically the general requirements for an ER in section 51.45. As noted in 10 C.F.R. § 51.41, the NRC requires an applicant to submit an ER to assist the Staff in complying with section 102(2) of NEPA, which identifies the broad issues to be addressed by a federal agency. Thus, the ER is a tool, together with other information, used to assist the agency in developing its EIS. Section 193(a) of the AEA and 10 C.F.R. § 51.20(b)(10) require the preparation of an EIS for any proposal to construct and operate a uranium enrichment facility.

The Staff evaluated the adequacy of the ER through a detailed technical review performed by the Staff and subject matter experts at ICF, Inc. Based on these reviews, the Staff requested additional information from USEC for both the draft and final EIS as outlined in 10 C.F.R. Part 51, Appendix A, and NUREG-1748, "Environmental Review Guidance for Licensing Actions Associated with NMSS Programs" (2003). In addition to written requests, the Staff held several teleconferences to clarify and obtain necessary information subsequently incorporated into revisions of the ER.

In addition to detailed technical review, the EIS process includes many other steps. After submittal of the ER, the Staff and subject matter experts at ICF began their review. This review was augmented by a public scoping meeting with members of the public, for the purpose of alerting the NRC to important areas and to provide additional information. Immediately prior to the scoping meeting, the Staff and ICF conducted a site visit to further familiarize themselves with various aspects of the proposed ACP site. Additionally, the Staff began consultations under Section 106 of the National Historic Preservation Act and Section 7 of the Endangered Species Act. The Staff and ICF used the information from the scoping meeting to focus the review and prepare the first set of RAIs issued in February 2005. This was an iterative process with the Applicant providing responses in numerous submittals. In addition to the ER, the Staff followed disposition of the interveners' contentions and issues in the LES hearing and included these areas within its review to the extent they were relevant to the proposed ACP. The Staff further augmented its review with additional RAIs pertaining to impacts from enrichment to 10%. After this initial review was complete the Staff issued the Drafts (DEIS)DEIS and held a public meeting to accept comments and gain further information. The Staff and ICF then reviewed all comments on the DEIS, revisited their analyses, completed consultations under Section 106 of the National Historic Preservation Act and Section 7 of the Endangered Species Act and subsequently issued the FEIS. Therefore, when looking to the adequacy of USEC's ER, one must consider all the other elements that go into preparation of the FEIS which ultimately serves in aggregate as the Staff's independent assessment.

B. Illustrate how machine cooling water systems work; show how this system assures there will be no water contaminated by uranium; and, discuss the anticipated success with achieving this result as compared to core cooling in a Pressurized Water Reactor (SER at 9-5).

Staff Response: As provided in Section 1.1.5.6.3 of the Application, the Machine Cooling Water (MCW) system is a closed-loop circulating water system designed to provide continuous cooling of the centrifuge diffusion pumps, lower suspension and drive assemblies, and the purge vacuum and evacuation from vacuum pumps. The system contains circulating water pumps, filters, heat exchangers, expansion tanks, and piping tie-ins to the chemical feed, deionizer, and sanitary water systems. Heated MCW leaves the centrifuge cascade through the service module header to an expansion tank, which provides enough suction head for the MCW circulating water pumps. The tank provides a convenient point for adding make-up water and water treatment chemicals. The discharge of the circulating pumps passes through an MCW filter and a heat exchanger where the MCW is cooled. The heat exchanger cooling water is supplied from a closed-loop Chilled Water (CW) system, and the CW chiller (heat exchanger) cooling water is supplied from the cooling tower and Tower Water Cooling (TWC) pumps. The cooled MCW then returns to the centrifuge machines by way of the supply header in the service module.

Sanitary water is provided for the MCW make-up water and the CW closed-loop. This water passes through a deionizer before entering either the MCW closed-loop or the CW closed-loop. The make-up water is used for initial fill purposes and for maintaining the proper level of MCW and CW in the system. MCW system alarms are monitored in the Area Control Room. Leakage from the MCW system and incidental spills of water elsewhere in the ACP are collected by the Liquid Effluent Collection (LEC) system. The LEC system consists of a set of drains and underground collection tanks for the collection and containment of leaks and spills of chemically treated water. The drains are located throughout the ACP. The tanks have a capacity of 550 gallons each and are monitored by liquid level gauges mounted above grade on pipe stands. Water accumulated in the LEC tanks is sampled and analyzed prior to disposal. If the contents meet the requirements of 10 C.F.R. § 20.2003, they may be pumped to the reservation sanitary sewer system. Otherwise, the tank contents will be containerized for off-site disposal.

As part of its environmental review, the Staff did not compare the MCW system leakage potential with that of a core cooling of a Pressurized Water Reactor (PWR) and has no regulatory basis to perform such a review, since the system is designed to be closed and any leakage is collected and sampled and handled in an appropriate manner. The MCW system is not a safety system, but rather a system that supports the efficiencies of the centrifuges. There is little likelihood that the MCW system would become contaminated with uranium since the system is sealed (i.e., closed-loop). Moreover, the MCW system is a positive pressure system that provides cooling or heat removal services to various components that support UF_6

enrichment processes or directly perform UF_6 enrichment. All UF_6 processing that is supported by the MCW system is at negative pressure; therefore, no UF_6 can migrate into the MCW system.

In any event, there is no comparison to be made with PWR core cooling systems because the design and operating parameters are so different. A PWR's coolant systems are operating at substantially greater pressure levels and there is no distinct pressure boundary separation between the reactor fuel and the coolant as there is between the MCW channels on the centrifuge casing and the vacuum conditions of the UF_6 inside the centrifuge casing. Also, core cooling systems become contaminated due to fuel tube failures and exposure to high neutron flux conditions present in a power reactor core that are not even conceivable in low level gas centrifuge enrichment processing conditions.

S8. Decommissioning

Chapter 10 of the SER addresses the NRC Staff's review of USEC's decommissioning plan. The Board has the following questions relating to the Staff's review of whether the site can be decommissioned safely and in accordance with NRC requirements.

A. Define what "promptly" means in regards to decontaminating and removing materials after cessation of operations (SER § 10.3.1.1).

Staff Response: "Promptly" means to decontaminate or remove equipment, structures, and portions of a facility and site containing radioactive contaminants to a level that permits the license to be terminated shortly after the cessation of operations. USEC intends to comply with the decommissioning timeliness requirements in 10 C.F.R. §§ 40.42(d) and 70.38(d) and not delay decommissioning activities after the cessation of operations. The decommissioning timeliness regulations in 10 C.F.R. §§ 40.42(d) and 70.38(d) establish requirements for notifying the NRC of pending decommissioning actions and cessations of licensee operations; establish requirements for when decommissioning plans need to be submitted; and establish requirements for completing decommissioning activities. In accordance with the decommissioning timeliness requirements in 10 C.F.R. §§ 40.42(d) and 70.38(d), USEC will be required to prepare and submit a decommissioning plan to the NRC at least 12 months prior to the expiration of the NRC license, and would begin the decontamination and decommissioning activities upon NRC approval of the final decommissioning plan. See SER at 10-3 and FEIS at 4-90. The decommissioning plan must include, among other things, a schedule for remediation activities and license termination. See 10 C.F.R. § 70.38(g)(4) and Section 5.1, NUREG-1757, Volume 1, “Consolidated NMSS Decommissioning Guidance - Decommissioning Process for Materials Licensees.” For decommissioning plans calling for completion of decommissioning later than 24 months after plan approval, a justification for the delay based on the criteria in 10 C.F.R. § 70.38(i) must be provided. USEC has estimated that decommissioning of the ACP will require approximately 6 years. See SER at 10-3.

B. Show where and how decontamination will take place at the ACP; illustrate which parts of the plant will be considered contamination zones; describe the specific good housekeeping practices that are required to maintain the areas outside of the contamination zones of the facility containment free; and, discuss how the Staff will assure that practices are implemented and performed (SER § 10.3.1.2).

Staff Response: Tables C3.4, C3.5, and C3.5(A) of the USEC Decommissioning Funding Plan list the areas and equipment expected to be contaminated. Decommissioning Funding Plan at pages C-3 and C-5 through C-9. Contamination would occur through contact of equipment and building surfaces with licensed material during plant operations. Actual contamination levels and locations will be described in the decommissioning plan that is required to be submitted at the end of operations in accordance with 10 C.F.R. §§ 40.42(d), 40.42(g)(4)(i), 70.38(d), and 70.38(g)(4)(i).

Decontamination is generally described in Section 10.8 of the USEC Application. Because casings for the cascades are in a vacuum, contamination of building and equipment surfaces is expected to be low. Contaminated process equipment will be removed using standard decommissioning techniques (e.g., unbolting flanged connections, cutting using portable saws, etc.). Equipment may be decontaminated using citric acid solution baths and degreasing. Classified components will be dispositioned appropriately. The Applicant has used these procedures in normal maintenance and cleanup activities at its GDPs. Specific procedures to be used in the decommissioning will be described in the decommissioning plan that is required to be submitted in accordance with 10 C.F.R. §§ 40.42(g)(4)(ii) and 70.38(g)(4)(ii) at the end of operations.

Contamination zones are shown in Appendix B, Figure 10.1-1, of the USEC Application. The Applicant proposed to use good housekeeping practices to minimize contamination during operations at the facility. These practices are generally described in Section 10.1.1 of the USEC Application. These practices include sectioning off clean areas, establishing buffer areas between clean areas and contamination zones, and applying standard radiation protection methods, such as purging contaminated equipment prior to removal, use of protective equipment, and establishing local ventilation controls. The NRC Staff will evaluate USEC's housekeeping practices as part of its initial operational readiness review required under 10 C.F.R. §§ 40.41(g) and 70.32(k) and routine radiation protection inspections during operations. These inspections will include reviews of procedures and how these procedures are implemented during operations.

C. Describe the facilities, procedures, and expected results of decontamination of the ACP that the Staff reviewed in determining the adequacy of the proposed plan (SER § 10.3.1.8).

Staff Response: Staff reviewed the general, programmatic descriptions of the facilities, procedures, and results of decontamination efforts presented in Section 10.8 of the Application. Regarding expected results, the Applicant is proposing to decommission to levels meeting the unrestricted use requirements in 10 C.F.R. § 20.1402. Specific, detailed information in the above areas is not required by the regulations at this time, however, the details will be described in the decommissioning plan that is required to be submitted in accordance with 10 C.F.R. §§ 40.42(g)(4) and 70.38(g)(4) at the end of facility operations.

B. <u>ENVIRONMENTAL ISSUES</u>

With respect to issues relating to the NRC's obligations under the National Environmental Policy Act (NEPA), the Board has been directed by the Commission to determine "whether the review conducted by the NRC Staff pursuant to 10 C.F.R. Part 51 has been adequate." In addition, the Commission directed the Board to make an independent determination regarding three "baseline NEPA issues." To assist the Board in making the required determinations, the Staff shall address the following issues, as well as provide any other background or supporting material that it believes will assist the Board in making its findings with regard to environmental/NEPA matters.

E1. The Board must independently consider the final balance among conflicting factors in the record and to determine whether the license should be issued, denied, or conditioned to protect environmental values. Based on the record to date, list all the conflicting factors that the NRC Staff believes should be balanced with regard to determining the appropriate action to be taken and provide detailed justification (with specific references to the record) to the Board for reaching a positive finding on the license decision.

Staff Response: There are six places in the FEIS that summarize and weigh the environmental factors that have a bearing on the license decision:

- The Executive Summary summarizes the potential environmental impacts of the proposed action (pages xxi-xxvii), the costs and benefits of the proposed action (pages xxvii-xxviii), and the tradeoffs associated with the no-action alternative (pages xxviii-xxix);
- Section 2.3 (pages 2-36 to 2-49) summarizes the environmental and other factors that led to several alternatives being eliminated from detailed consideration, including the environmental tradeoffs of siting the ACP at Piketon versus the Paducah DOE reservation (Table 2-7 on pages 2-38 to 2-42);
- Table 2-8 (pages 2-49 to 2-61) summarizes and compares the environmental impacts for the proposed action and the no-action alternative;

- Section 2.5 (pages 2-61 to 2-62) provides the Staff recommendation regarding the proposed action based on our assessment of potential environmental impacts;
- Chapter 7 provides an overall cost benefit analysis, including a comparison of the costs and benefits associated with the proposed action (Section 7.1 on pages 7-1 through 7-5) and a comparison of the costs and benefits for the proposed action relative to those of the no-action alternative (Section 7.2 on pages 7-6 to 7-10); and
- Chapter 8 summarizes the environmental consequences of the proposed action, including unavoidable adverse environmental impacts (Section 8.1 on pages 8-1 to 8-3), the relationship between local short-term uses of the environment and the maintenance and enhancement of long-term productivity (Section 8.2 on pages 8-3 to 8-4), and the irreversible and irretrievable commitment of resources (Section 8.3 on page 8-4).

Section 7.1 provides perhaps the best single comparison of the costs and benefits that should be balanced when determining the appropriate action. This comparison is supported by more detailed analyses that are referenced in other sections of the FEIS. All of these conflicting factors and the Staff's resulting conclusions are summarized below.

Costs of the Proposed Action

- Adverse Environmental Impacts at Piketon. As summarized on pages 7-2 through 7-4 (and other sections in the FEIS referenced above), the proposed action would result in small impacts to land use, historical and cultural resources, visual and scenic resources, air quality, geology and soils, water resources, ecological resources, noise, and public and occupational health. There would be no environmental justice concerns. Transportation impacts, while generally small, could reach moderate levels resulting from hypothetical accidents with a low probability and severe consequences (i.e., an accident releasing large quantities of UF_6 as discussed in Section 4.12.2.1. In addition, routine waste management impacts would be small, but the ACP would significantly increase the onsite inventory of depleted UF_6 stored at Piketon and would require DOE to significantly extend the life of the UF_6-to-U_3O_8 conversion facility at Piketon beyond its 18-year planned lifespan.

- Indirect Costs to the Local Economy Around Piketon. As discussed in Section 4.2.8 starting on page 4-29 of the FEIS, the socioeconomic impacts in the region of influence would include impacts to area housing resources, community and social services, and public utilities. These impacts are estimated to be small.
- Adverse Socioeconomic Impacts Associated with the Cessation of Operations at the Paducah Gaseous Diffusion Plant. As discussed in Section 4.2.8.4 starting on page 4-36 of the FEIS, ceasing operations at Paducah would substantially reduce the number of full-time workers from current levels of approximately 1,870 full-time employees. These losses would be temporarily mitigated to some extent by the hiring of decommissioning workers in the event that the Paducah plant is decontaminated and decommissioned. Ceasing operations at Paducah would also result in small impacts to the local tax revenue, population size, area housing resources, community and social services, and public utilities.
- Direct Costs Associated with Proposed Action Life-cycle Stages. Site preparation and construction would cost an estimated $1.4 billion between 2006 and 2010, centrifuge manufacture and equipment assembly would cost an estimated $1.4 billion between 2004 and 2013, and decontamination and decommissioning would cost an estimated $516 million between 2040 and 2045. There would be additional direct costs associated with facility operation between 2010 and 2040, which are withheld from the FEIS because they are considered proprietary. See Table 7-1 on page 7-2 for more detail.

Benefits of the Proposed Action

- Provide Enriched Uranium to Fulfill Domestic Electricity Requirements. As discussed in Section 1.3.1 on pages 1-3 to 1-5 of the FEIS, the demand for enriched uranium to meet growing electricity requirements in the U.S. is on the rise, while the supplies of enriched uranium currently used in the U.S. are on the decline. Even with the recently licensed LES Facility and with the ACP operating at its potential full 7 million SWU capacity, the U.S. demand for enriched uranium will exceed the supply.
- Improve National Energy Security by Providing Increased Domestic Supplies of Enriched Uranium. As discussed in Sections 1.3.1 and 1.3.2 on pages 1-3 to 1-6 of the FEIS, up to 86 percent of the U.S. demand for enriched uranium is currently supplied by foreign sources. Even with uranium enrichment services from the recently-licensed LES Facility, the U.S. will remain dependent on foreign sources for most of its enriched uranium needs. The ACP would provide up to an additional 7 million SWU of domestic production capacity, increasing supplies from within the U.S. while allowing USEC to also retire the aging Paducah plant.
- Upgrade Uranium Enrichment Technology in the U.S. As discussed in Section 1.3.3 on page 1-6 of the FEIS, the gas centrifuge technology is known to be more efficient and require less energy and other resources to operate than the gaseous diffusion technology currently in use. The energy requirements of a gas centrifuge plant are estimated to be about 5 percent of that required by a comparably sized gaseous diffusion plant.
- Positive Socioeconomic Impacts in the Region Around Piketon. As discussed in Section 4.2.8 starting on page 4-29 of the FEIS, the proposed action would result in small positive socioeconomic impacts in the region around Piketon. Direct and indirect

jobs (annual, full-time jobs) created by the proposed action would include an estimated 3,362 jobs during site preparation and construction between 2006 and 2010, and 1,500 jobs during ACP operations between 2010 and 2040. There would also be an increase in tax revenues.

Staff Conclusions: As noted in Section 7.3 of the FEIS the Staff believes that the benefits of the proposed action significantly outweigh the environmental and socioeconomic costs of the proposed action. We therefore believe that the balance of these factors supports issuing the license to USEC.

E2. In regards to the accident analysis in FEIS Appendix H (withheld pursuant to 10 C.F.R. § 2.390), verify that this is the same analysis presented in Appendix B of the SER and:

A. Provide a discussion of the accident analysis which presents as the most severe accident with regard to the public health and safety.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

B. Describe the emergency plan outlining mitigating actions that could be taken to reduce the consequences of that accident, presenting an example of actions that could take place in the area affected by the accident.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

C. Provide the Board with information regarding what other mitigating actions are potentially available to reduce the consequences of that type of accident.

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

E3. In regards to determining the extent of the NRC Staff's review relating to FEIS Chapter 3 (Affected Environment), Chapter 4 (Environmental Impacts), and Chapter 6 (Environmental Measurement and Monitoring Programs), clarify the Staff's understanding of the proposed monitoring program as it relates to existing and future impacts and the relationship between DOE and ACP monitoring by addressing the following:

A. FEIS at 4-24 to 4-25:

1. While the proposed groundwater extraction is projected to be only 31% of the design capacity and permitted rate of the well field groundwater withdrawal system, has the well field ever been pumped at its permitted rate to verify its capacity?

Staff Response: Three well fields (X-608, X-605, and X-6609) in the Scioto River Aquifer having a total design capacity of 34 million gallons per day (gdp) provide water to the site via the X-611 water treatment plant, which has a design capacity of 20 million gdp. Thus, the effective design capacity of the system to provide water to the site is 20 million gdp. However, the fields are currently producing 5.5 million gallons per day. Prior to entering the cold standby mode in May 2001 the average daily water use for the Portsmouth plant reported by DOE in the Annual Site Environmental Report, was 13 million gallons per day (combined for the three well fields). Current use is 5.5 million gallons per day. The additional water use from the ACP will increase the average daily water use to 6.15 million gallons per day. Thus the ACP will result in water use at 47% of the average daily use during the operation of the Portsmouth plant.

2. Summarize the Staff's evaluation of the design features that address potential groundwater impacts from breached cylinders of DU in the storage yards.

Staff Response: The cylinders primarily used for storage of tails (solidified depleted UF_6) are known as Model 48G cylinders. These cylinders are made of carbon steel and are about 4 feet in diameter, 12 feet long, and weigh about 30,000 pounds when full. Model 48G cylinders have been determined to meet the requirements for a DOT-7A, Type A packaging per 49 C.F.R.§ 178.350. Details on the testing and evaluation conducted on these cylinders can be found in "Test and Evaluation Document for the U.S. Department of Transportation Specification 7A Type A Packaging," (DOE/RL-96-57, Rev. 0, Volume 1). In Chapter 1 of the Application, the Applicant has committed to follow ANSI N14.1-2001, "Nuclear Materials –

Uranium Hexafluoride - Packaging for Transport," in order to ensure the integrity of the cylinders containing UF_6.

A filled cylinder is not moved to a cylinder yard (where it is stacked in place) until the depleted UF_6 is solidified. The yards are constructed with sealed airport runway-quality concrete. All of the cylinder storage yards are designed primarily for storage of 2.5, 10, and 14-ton UF_6 cylinders. The cylinders would be inspected and maintained while being stored onsite. Maintenance activities would include periodic inspection for corrosion, valve leakage, or distortion of cylinder shape. Repainting of the cylinders would be conducted as indicated by the inspections. Depleted UF_6 may be transferred into new cylinders during plant operation in the event that cylinder inspection indicates potential loss of cylinder containment.

Because the UF_6 cylinders would be stored on an impervious surface, would be maintained and inspected routinely, and all runoff water would be collected and directed to an existing holding pond that is routinely analyzed for radionuclides, soil and geology would not be impacted by the operation of a storage yard. The routine maintenance and inspections would identify and correct any potential or actual releases, and any releases transferred via runoff would be detected in the holding pond and remediated.

Based on these factors, and the results of accident analyses discussed in the SER, the Staff believes that USEC has adequately addressed the potential impacts to groundwater in the case of a breached cylinder.

Related information is provided in the Staff's response to E5.B.

3. What is the percentage increase over the maximum historic withdrawal rate from each of the two well fields (FEIS at 8-4)?

Staff Response: The historic withdrawal rates for the three (not two) well fields are not reported. Prior to entering the cold standby mode in May 2001, the average daily water use for the Portsmouth plant reported by DOE in the Annual Site Environmental Report, was 13 million gallons per day. Current use is 5.5 million gallons per day. The additional water use from the ACP will increase the average daily water use to 6.15 million gallons per day. The action will result in water use at 47% less than the average daily use during the operation of the Portsmouth plant.

B. FEIS at 6-6:

1. Verify that the underground liquid storage tanks that collect leaks and spills of treated water are the only underground facilities at ACP, exclusive of the associated drains and interconnecting piping.

Staff Response: The facility descriptions provided by the Applicant in the Environmental Report do not state unequivocally that there are no other underground systems. It is reasonable to assume that other underground systems such as service water piping, sewage, etc. exist. However, with respect to additional waste collection or treatment components located underground (such as pumps for the tanks), there are no other underground collection systems in existence of these facilities other than the Liquid Effluent Collection (LEC) System.

2. Clarify what is meant by "routine" monitoring and "tracking the levels" for these tanks.

Staff Response: We understand that the Applicant `has some procedures for their existing tank monitoring programs that it will adapt for the ACP monitoring program. These procedures should describe the parameters being monitored and the monitoring frequencies.

Daily, each 12 hour shift. Operations monitors the gauge reading for each tank. When tank levels reach 500 gallons the tank content is sample and analyzed. Analytical results for LEC waters are reviewed by Environmental Compliance to determine allowable disposal options. Upon approval of a disposal method by Environmental Compliance, the LEC tank(s) are pumped out. If the LEC tank content is acceptable for disposal at the X-6619, NPDES permitted discharge, the water will be discharged to the sanitary sewer. If the LEC tank content is not acceptable for the X-6619, the water may be treated at the X-705, through a service contract, with the consent of X-705 management and then discharged to the X-6619

3. Discuss the effectiveness of level gauges in detecting small leaks and weeps from the drain tanks.

Staff Response: The LEC system is intended to collect leaks and spills from the recirculating heating water, machine cooling water, tower cooling water, and fire protection water systems through a system of floor drains, piping, and underground collection tanks that are designed to catch and collect liquids spilled within the ACP buildings. Also collected will be mop water from floor cleaning operations and oily water from the generator rooms, and vacuum pump stations. The system is not intended to collect routine effluents other than inadvertent mop water but is intended to collect leaks and spills.

The LEC system consists of a system of floor drains, which are connected by underground piping to 550-gallon fiberglass single walled collection tanks buried outside the buildings. The LEC tanks are required to be inspected and tightness tested once every three years.

4. Is there any potential for leakage from pipelines, valves, or other similar facilities to bypass the Liquid Effluent Control System, seep into the ground, and impact groundwater at the site? If so, how will this inadvertent release be monitored and remediated? (FEIS at 4-23).

Staff Response: The process equipment that handles liquid radioactive material is located in buildings and facilities that are covered by the LEC system. Leakage from these systems should be captured by the LEC system. The LEC system consists of a system of floor drains which are connected by underground piping to 550-gallon fiberglass single walled collection tanks buried outside the buildings. The LEC tanks are required to be inspected and tightness tested once every three years.

Daily, each 12 hour shift. Operations monitors the gauge reading for each tank. Should a discrepancy in the gauge reading (mass balance) be determined, an investigation will be conducted to determine the cause of the discrepancy.

5. Explain how the impacts from any ACP releases to surface water as measured by the outfalls will be separated out from the impacts from other activities at Portsmouth.

Staff Response: As discussed in Section 4.2.6.2 starting on page 4-20 of the FEIS, sanitary wastewater, discharges from the tower water cooling system, and storm water runoff from the ACP would flow though existing onsite treatment systems and be discharged through existing NPDES-permitted outfalls. Monitoring at those outfalls would ensure that the commingled wastewaters met permitted release levels, but would not determine what fraction of measured concentrations was the result of ACP releases versus other site releases. However, given the ACP's process, we expect the relative contribution of radionuclides to these liquid waste streams from the ACP to be very small and the combined releases to be within 10 C.F.R. Part 20 release limits.

Separate from the above three liquid waste streams, the Applicant would contain and conduct separate monitoring of any leakage from the machine cooling water system and any incidental spills of water elsewhere in the ACP. As discussed in Section 4.2.6.2, these liquids would be collected by the LEC system, consisting of a set of drains located throughout the ACP and underground collection tanks. The Applicant would sample and analyze the water accumulated in the Liquid Effluent Collection tanks prior to disposal. If the contents meet the requirements of 10 C.F.R.§ 20.2003, they may be pumped to the reservation's sanitary sewer system and discharged along with treated sanitary wastes from other onsite activities. Otherwise the tank contents would be containerized for offsite disposal. We believe this proposed containment and monitoring would provide an appropriate basis for determining and controlling the impacts associated with ACP liquid discharges most likely to contain radioactive material.

6. How will the monitoring commitments listed in the last paragraph of FEIS section 6.1.3 be documented in the license?

Staff Response: These monitoring commitments can be found in Section 4.4.3.1 of the ER. The standard practice for licensing fuel cycle facilities is to directly incorporate license application documents, including the ER, into the license by tie-down references. In this way, all the Applicant's commitments documented in the licensing documents become enforceable, subject to the 10 C.F.R. § 70.22 change process.

E4. In regards to the impact from the ACP operations (FEIS § 4.2.12.3):

A. For routine radiological impacts, what is the basis for the NRC Staff's conclusion that 10% enrichment would be infrequent? Summarize the assessments (discussed in FEIS at 4-62 to 4-65) that were performed for 10% enrichment. If none were performed, what control does the NRC have to limit the amount of 10% enrichment so that it would not be the controlling enrichment level for airborne releases?

Staff Response: Based on discussion with USEC, the Staff believes that 10% enrichment would be infrequent, particularly in the early years of ACP operation. Current generation nuclear power plants do not use enrichments above the original requested enrichment limit of 5% (the customer product range is typically from approximately 2.4 percent to 5 percent), but a potential exists for either research reactors to request enrichments above 5% or for future generation nuclear power plants to utilize enrichments above 5% (e.g., the approximately 8% expected for Modular Pebble Bed Reactors). Since these possible future business opportunities may materialize, the Applicant has indicated that they would prefer to get approval now for 10% operations rather than be limited in their future business flexibility by the process of modifying their license to permit 10% operations. Accordingly, given the uncertainty concerning when enrichment activities might increase above 5% assay, and the fact that enrichment will likely remain below 5% in the early years of ACP operation, the staff summarized these conditions as "infrequent." Notwithstanding the terminology, analysis was performed at 10% assay to ensure the results would meet regulatory requirements, should the Applicant enrich up to 10% assay.

The 10% enrichment analysis assumed that a constant mass of uranium would be released from each of the listed facilities, regardless of whether that uranium was from a 5% enrichment process or a 10% enrichment process. The mass of each uranium isotope released from each facility was then calculated by multiplying a facility-specific isotopic mass fraction times the total mass of uranium expected to be released from that facility. The facility-specific isotopic mass fraction was calculated based upon the average enrichment of the uranium that would be handled at that facility. For example, during enrichment to 10%, the process buildings would be expected to contain uranium with an average enrichment of about 5%, the feed facility would be expected to handle natural uranium, and the customer services facility would be expected to handle uranium that has been enriched to 10% U-235. All other analysis parameters such as weather data, receptor locations, etc., were identical to those used in the analysis for the 5% U-235 enrichment scenario.

The isotopic mass fractions were developed for U-234, U-235, and U-238 using equations that calculate product isotopic mass fractions as a function of U-235 enrichment for the centrifuge process (also known as separation factors). The centrifuge process generates a high separation factor for U-234, with the mass concentration of U-234 increasing faster than the concentration of U-235 in the product. By assuming that the ACP operates in a 10% enrichment mode for the entire year, the calculation maximizes the mass fraction of U-234 in the released uranium. The high specific activity of U-234 then produces a conservatively high dose relative to the expected operation scenario of only short runs in 10% enrichment mode over the course of a year. Despite this conservatism in the analysis, the resultant maximum dose of

1.1 mrem per year is still well below applicable regulatory limits. Although the resulting maximum dose is below the applicable regulatory limits, no additional control by NRC is necessary to limit the amount of enrichment because the Applicant would still have to meet the regulatory requirements and limit its operations accordingly.

B. Is the X-710 Laboratory Facility part of the ACP, and where is it located on Figure 2-4 (FEIS at 2-8)? The discussion in the FEIS at page 4-64 seems to imply that the reported air emissions include all the facilities at Piketon. Is this correct, and if not, why?

Staff Response: The X-710 Laboratory is located just east (across the street) of the midpoint of the X-326 building. The X-326 Building is the southernmost of the two large GDP buildings aligned on a north-south axis near the middle of the site. The FEIS considers operational radiological air emissions from only those radiological facilities expected to be used as part of ACP operations. USEC did not propose the X-710 Laboratory as one of the facilities that would be part of the ACP, but stated that the ACP would require analytical services and the existing United States Enrichment Corporation X-710 Laboratory would be an obvious potential supplier of those services. Therefore, as enumerated on page 4-63, the X-710 Laboratory was among the site facilities included in our analysis. Radiological emissions from other onsite facilities that are not related to the operation of the ACP were not included in the FEIS airborne release analysis.

C. Relating to impacts from plausible accidents (FEIS at 4-71):

1. What design features are proposed to prevent a “domino” effect associated with multiple centrifuge failures initiated by the structural failure of one cylinder?

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

2. Is there any redundancy in the high temperature and high pressure trips to protect the centrifuges from over-pressurization breaches?

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

3. Clarify the potential for nuclear criticality that has been evaluated for the 10% enrichment.

Staff Response: Nuclear criticality is considered a high-consequence event for purposes of the ISA. This is based on the potential to exceed an exposure of 100 ram to workers (10 C.F.R. § 70.61(b)(1)), and therefore criticality must be made highly unlikely under 10 C.F.R. § 70.61. Criticality accidents have historically resulted in significant exposures only to individuals close to the reacting material (e.g., lethal doses typically occur within approximately 15 feet), so that off-site consequences to members of the public and the environment are relatively minor. The generic criticality accident is intended to demonstrate the approximate level of off-site exposure (the vast majority of which would be due to migration of the fission-product cloud and not to direct radiation). However, the practice of preventing criticality accidents to be highly-unlikely for the purpose of limiting dose to workers ensures that the risk to members of the public and the environment is bounded. At an enrichment of 10% U-235, the consequences are not significantly greater than would occur at 5% U-235. The landmark report LA-13638, "A Review of Criticality Accidents," contains details on 22 process criticality accidents between 1953 and 1999. These have included accidents in uranium systems with enrichment between 6.5 and 93% U-235, as well as plutonium systems. There does not appear to be any noticeable correlation between enrichment and the amount of energy released. The significance of the higher enrichment is that it would take smaller masses, dimensions, and levels of moderator to achieve a critical configuration than at lower enrichment. Therefore, while the likelihood of criticality would be greater at higher enrichment, consequences would not necessarily be worse.

E5. Options to mitigate adverse environmental impacts for the proposed action presented in FEIS Chapter 5 raise the following concerns:

A. Explain how the mitigative measures proposed in Tables 5-1, 5-2, and 5-3 (FEIS at 5-2 to 5-4) will be incorporated into the license and how they will be implemented, monitored and evaluated during construction and operations.

Staff Response: These mitigation measures can be found in Sections 1.3, 4.3.3, 4.4.3, 4.5.3, 4.12.3, and 5.0 of the ER. The standard practice for licensing fuel cycle facilities is to directly incorporate license application documents, including the ER, into the license by tie-down references. In this way, all the Applicant's commitments documented in the licensing documents become enforceable. Specific details regarding implementation of these mitigation measures were not provided in the ER. Monitoring and evaluation will be performed through inspections conducted by the Region as discussed in the answer to Question S2-1. A.6.

B. Explain in more detail the proposed visual inspections of the cylinder storage yards in regards to their potential impact on geology and soils (FEIS Table 5-2).

Staff Response: This response contains information designated Official Use Only-DOE NOFORN and is included in Appendix A.

E6. The Board has the following questions on the cost benefit analysis in Chapter 7:

A. Elaborate on the basis for generating the cost benefit figures presented in Table 7-1 (FEIS at 7-2) and Table 7-2 (FEIS at 7-5).

Staff Response: The sources and methodologies for the cost benefit analysis presented in Chapter 7 are extensively discussed in Appendix G of the FEIS.

In Table 7-1, costs and benefits raw data were sourced from:

- (USEC, 2005a) USEC Inc. "Environmental Report for the American Centrifuge Plant in Piketon, Ohio." Revision 6. Docket No. 70-7004. November 2005.
- (USEC, 2005b) United States Enrichment Corporation. "Additional Responses to Request for Additional Information Regarding the Environmental Report (TAC No.L32307) - Proprietary Information." Dated April 21, 2005.

In Table 7-2, data on socioeconomic impacts were sourced from:

- (USEC, 2005) USEC Inc. "Environmental Report for the American Centrifuge Plant in Piketon, Ohio." Revision 6. Docket No. 70-7004. November 2005.

The Applicant generated the socio-economic impact estimates listed in Table 7-2 using the RIMS-II input-output model. Input-output models are a standard tool in economic analyses for estimating the aggregate impacts – such as indirect jobs created and tax revenues generated -- of project expenditures in specific regions of the country. The RIMS-II model was developed by the U.S. Department of Commerce's Bureau of Economic Analysis (BEA) and has been used extensively in public sector and private economic analyses. The model requires assumptions about the number of direct jobs created, project-related expenditures, and regional input-output multipliers. ICF verified the accuracy of the input parameters in the Applicant's application of the RIMS-II model and the reasonableness of the output variables.

B. What assurances are there that the enriched uranium will be available for domestic use and not diverted to foreign markets (FEIS at 7-4)?

Staff Response: There are no statutory or regulatory requirements prohibiting the sale of enriched uranium produced in the U.S. for peaceful use in foreign countries. Such exports are authorized under the AEA in accordance with Agreements for Cooperation entered into by the U.S. Government under Section 123 of the Act. Section 193 of the AEA prohibits the issuance of a license to the Applicant if such issuance would be "inimical to … the maintenance of a reliable and economical domestic *source* of enrichment services." The ACP will clearly provide a new domestic source of such services. It will be constructed in the U.S. using U.S. technology and will be owned and operated by a U.S. company that is not foreign owned, controlled or influenced. The ACP will serve both the domestic and foreign markets. In 2005 the Applicant's estimated market share of the SWU component of low enriched uranium purchased by and shipped to utilities in North America was 53% and its estimated market share in the world was 27%. No NRC regulation prohibits foreign sales. As discussed in the FEIS, licensing of the ACP expands and diversifies U.S. domestic sources of enriched uranium. FEIS at 7-7,

C. Now that the Louisiana Enrichment Facility (LES) is licensed and under construction, explain what effect it will have on the no-action alternative for the ACP (FEIS at 7-6 to 7-10).

Staff Response: Licensing of the LES facility will have no effect on the no-action alternative evaluated in the FEIS. The no-action alternative explicitly addressed the anticipated licensing of the LES facility. FEIS at 2-36. In addition, as discussed on pages 1-4 and 1-5 of the FEIS, with or without the LES facility, the no-action alternative would result in a shortfall in U.S. domestic supply of enriched uranium against U.S. domestic demand. The LES facility would satisfy only about 25% of the U.S. demand. The ACP is intended to help offset the eventual cessation of operation of the Paducah GDP and the U.S. reliance on the Megatons to Megawatts program which relies on foreign sources of uranium. The Megatons to Megawatts program is scheduled to end in 2013.

E7. In addition to the environmental issues above, please address the following:

A. Based on current thinking, outline and discuss the components and processes envisioned for inspecting and monitoring the cylinders in the storage yards (FEIS at 2-30).

Staff Response: See response to questions E5.B.

B. Is the socioeconomic impact shown on Table 2-7 a positive impact rather than a negative impact? If so, is the greater impact for Paducah an advantage of the site over Piketon? (FEIS at 2-40).

Staff Response: As discussed in the “land use” resource area assessment in Table 2-7, locating the project at Paducah requires more construction activity than if the project were located at Piketon. As a result of the higher construction requirements, more employment would be generated at the Paducah location. It is not appropriate to judge the employment impact of the project alternatives in isolation. Lower construction costs at Piketon would result in lower costs of enriched uranium and consequently lower cost electricity, which in turn would increase economic competitiveness. Moreover, as stated in the FEIS, the Staff found that the selection of the Paducah site would result in somewhat greater environmental impacts due primarily to the need for construction of all new buildings, and the attendant excavation and land disturbance. FEIS at 2-37.

C. How does DOE's deconversion plant at Portsmouth manage and handle hydrofluoric acid, and why is this considered an extra burden if deconversion is performed at a fuel fabrication facility (FEIS at 2-48 to 2-49)?

Staff Response: DOE's plan for conversion of DUF_6, (as described in the "Final Environmental Impact Statement for Construction and Operation of a Depleted Uranium Hexafluoride Conversion Facility at the Portsmouth, Ohio Site." DOE/EIS-0360. Office of Environmental Management. (June 2004)), would utilize a continuous dry conversion process in which DUF_6 is vaporized and converted to a mixture of uranium oxides (primarily U_3O_8) by reaction with steam and hydrogen in a fluidized-bed conversion unit. The resulting depleted U_3O_8 powder would be collected and packaged for disposition. Each process line would consist of two autoclaves, two conversion units, a hydrofluoric acid (HF) recovery system, and process off-gas scrubbers. The facility would have three parallel conversion lines. Equipment would also be installed to collect the HF co-product and process it into any combination of several marketable products. A backup HF acid neutralization system would be provided to convert up to 100% of the HF acid to CaF_8 for storage, sale, or disposal in the future, if necessary.

Management of the HF generated from conversion would not be considered an extra burden if this activity were undertaken at an alternative site such as a fuel fabrication facility. Management of this material would be required regardless of whether the DUF_6 conversion occurs at the DOE facility, the Portsmouth facility, or at an existing fuel fabrication facility.

D. Clarify the following (see Table 2-8 (FEIS at 2-49):

1. Why is the potential for additional domestic enrichment facilities being constructed in the future is included in the no-action alternative?

Staff Response: The no-action alternative would result in the NRC not issuing a license for the construction, operation, and decommissioning of the ACP. The need for a reliable, domestic source of low-enriched uranium would still exist, however. One of the possible ways that this domestic source could be developed would be by licensing other domestic enrichment facilities in the future. Because the need for these other domestic enrichment facilities would possibly be reduced or eliminated with the licensing of the ACP, the Staff considered this scenario to be a function of the implementation of the no-action alternative as it relates to the licensing of the proposed ACP.

2. How can the "additional domestic enrichment facilities" for the no-action alternative have more impact than the ACP plant for Public and Occupational Health and for Waste Management (FEIS at 2-60 to 2-61)?

Staff Response: If any additional domestic enrichment facilities are proposed in the future, the environmental impacts at any alternate site(s) would have to be assessed in a separate NEPA review Sections 4.4.12 and 4.4.13 of the FEIS indicate that the impacts for public and occupational health and waste management are expected to be similar to those of the proposed action, i.e., SMALL. The impacts indicated in Table 2-8 should have stated that the impacts of the no action alternative should be SMALL for these resources. However, the construction and operation of another enrichment facility in the United States, needed to fulfill growing demands, could vary from the proposed action, depending on the particulars of the proposed action and ecological conditions at any alternate site(s). Those impacts would have to be evaluated in a separate NEPA review.

E. The FEIS states on page 3-6 that ground subsidence impacts from pumping from the well locations are considered in Section 4.2.6. However, Section 4.2.6 only has a conclusory statement that the Ohio EPA confirmed that subsidence and sinkholes from groundwater withdrawal are not an issue in the region (FEIS at 4-24). To clarify this in relationship to any potential historic feature (such as the possible Great Hopewell Road (see FEIS at B-229)) discuss the following:

1. What are the depths of the well screens and what geologic material are these wells developed in?

Staff Response: Water for USEC is provided by DOE and is obtained from three well fields (X-608, X-605, and X-6609) located within the Scioto River Valley. Well field X-6609 is operated by the United States Enrichment Corporation in the area of the possible Great Hopewell Road. The 12 production wells in this field were drilled in June 1986 and are arranged in a linear pattern along a relatively straight stretch of the Scioto River. The wells are located in the river floodplain between the river and Route 23. The surface of the wells may occasionally be under water when the river level is high. Details of the well construction were obtained from the Ohio Department of Natural Resources (DNR), Division of Water, well log database, and are summarized in the table below. All wells are screened in the Newark River Overwash, a sand and gravel aquifer created by glaciers. Well screens are 25 feet long and range in depth from 39 feet (top of screen) to 78 feet (bottom of screen) below the ground surface.

Well Screen and Geologic Information for Well Field X-6609

Well Number	Ohio DNR Log Number	Well Diameter (inches)	Top of Screen (Feet below ground surface)	Screen length (feet)	Bottom of Screen (Feet below ground surface)	Geologic Description
1	571217	16	51	25	76	Medium Sand and Gravel
2	571218	16	53	25	78	Fine and Medium Sand and Gravel
3	571219	16	53	25	78	Fine and Medium Sand and Pea Gravel
4	571220	16	47	25	72	Fine and Medium Sand and Gravel
5	571221	16	45	25	70	Medium to Fine Sand and Gravel
6	571222	16	45	25	70	Medium to Fine Sand and Pea Gravel

Well Number	Ohio DNR Log Number	Well Diameter (inches)	Top of Screen (Feet below ground surface)	Screen length (feet)	Bottom of Screen (Feet below ground surface)	Geologic Description
7	571223	16	44	25	69	Medium to Fine Sand and Pea Gravel
8	571224	16	46	25	71	Sand and Pea Gravel
9	571225	16	39	25	64	Medium to Fine Sand and Gravel
10	571226	16	50	25	75	Medium to Fine Sand and Gravel
11	571227	16	44	25	69	Medium to Fine Sand and Gravel
12	571228	16	46	25	71	Medium to Fine Sand and Gravel

All data from Water Well Logs and Drilling Reports are on file with the Ohio DNR, Division of Water.

2. What is the average annual and maximum pumping rates that have actually occurred in the well field since these wells were installed?

Staff Response: In the past, up to three well fields (X-605G – 4 wells; X-6609 – 12 wells; and X-608 A and B – 15 wells) supplied water to the site. Production from well field X-6609 began in 1988, and production from well field X-605G was terminated in 2000. Currently, well fields X-6609 and X-608 A and B supply water to the site at an average rate of up to 6.6 million gallons a day recorded in 2001. Prior to May 2001, when the Portsmouth facility and all three well fields were in operation, average daily water production varied from 9.9 million gallons per day in 1998 to 17.3 million gallons per day in 1978. The following table summarizes the average annual and maximum pumping rates associated with each well field since 1976.

Year	X-605G Well Field		X-6609 Well Field		X-608A and B Well Field		Total Average
	Max	Average	Max	Average	Max	Average	
1976	2.5	1.2	NP	NP	15.0	11.7	**12.9**
1977	2.6	2.1	NP	NP	16.9	15.0	**17.1**
1978	2.8	2.2	NP	NP	18.9	15.1	**17.3**
1979	3.4	2.4	NP	NP	16.4	11.8	**14.2**
1980	2.6	2.3	NP	NP	11.9	10.0	**12.3**
1981	2.4	1.5	NP	NP	12.0	8.5	**10.0**
1982	2.7	1.0	NP	NP	11.4	9.8	**10.8**
1983	1.3	0.3	NP	NP	11.1	9.7	**10.0**
1984	2.9	0.7	NP	NP	13.2	10.9	**11.6**
1985	2.6	1.1	NP	NP	17.6	13.2	**14.3**
1986	1.7	0.6	NP	NP	14.4	12.6	**13.2**
1987	1.2	0.4	NP	NP	15.5	13.2	**13.6**
1988	1.1	0.6	4.7	3.0	10.6	7.6	**11.2**
1989	2.5	1.7	6.1	4.8	11.2	8.5	**15.0**
1990	2.5	2.1	5.2	3.8	15.1	10.3	**16.2**
1991	3.1	1.2	4.3	3.5	12.6	10.1	**14.8**
1992	1.1	0.7	7.2	5.3	11.3	9.0	**15.0**
1993	0.7	0.2	7.8	6.2	7.4	6.2	**12.6**
1994	2.5	1.0	4.8	3.8	13.1	10.3	**15.1**
1995	2.3	1.5	4.4	3.5	13.1	9.9	**14.9**
1996	1.4	0.5	5.4	3.8	8.7	8.0	**12.3**
1997	0.9	0.7	5.1	3.5	7.7	6.8	**11.0**
1998	2.6	1.6	3.5	2.6	6.6	5.7	**9.9**
1999	2.9	1.1	4.0	3.2	8.8	6.3	**10.6**
2000	1.2	0.1	4.5	3.6	12.2	8.1	**11.8**
2001	NP	NP	4.2	2.5	5.0	4.1	**6.6**
2002	NP	NP	1.9	1.2	4.3	3.1	**4.3**
2003	NP	NP	2.3	1.9	4.2	2.9	**4.8**
2004	NP	NP	2.3	1.3	5.1	3.8	**5.1**
2005	NP	NP	2.1	1.2	3.2	2.1	**3.3**
2006	NP	NP	1.9	1.1	3.4	2.0	**3.1**

Notes:

1. NP = no production
2. Production from wellfield X-6609 began in calendar year 1988
3. Production from wellfield X-605G terminated during calendar year 2000

After wells in field X-6609 were constructed in 1986, pumping tests were performed. During the tests, the wells were pumped at a rate of 500 gallons per minute for 48 hours and the drawdown for each well was recorded (where drawdown is the increased depth to the static water level). Drawdown ranged from 1.9 feet to 10.5 feet. Static water level in these wells ranged from 9.8 to 20.9 feet below ground surface. The pump test data are summarized in the table below:

Pump Test Information for Well Field X-6609

Well Number	Static Water Level (Feet below ground surface)	Top of Screen (Feet below ground surface)	Drawdown (feet)
1	20.9	51	9.7
2	18.4	53	7.6
3	16.5	53	6.7
4	18.3	47	6.9
5	16.1	45	5.2
6	16.0	45	7.8
7	15.2	44	6.1
8	9.8	46	1.9
9	13.1	39	8.6
10	19.8	50	9.3
11	14.8	44	8.2
12	15.3	46	10.5

3. How much could this pumping rate increase with the operation of the ACP and what is the estimated increased drawdown, if any, with the proposed pumping rate?

Staff Response: The highest water usage occurred during operation of the Portsmouth plant prior to May 2001, when water production and use averaged 17.3 million gallons per day in 1978. The plant currently uses between 3 and 5 million gallons per day, which represents between 17 and 29 percent of that used when the plant was in full operation. The proposed ACP would use an additional 0.65 million gallons per day increasing the total site water use to up to 6.15 million gallons per day, or 36 percent of the water used when the Portsmouth GDP was in production mode. The additional water use for the proposed action would result in values significantly less than historical amounts and would not result in an increase in pumping or cause an increase in drawdown above historical use.

Studies by researchers at Ohio State University in 1994 showed that 50 to 88 percent of the well field water is drawn from the Scioto River (Nortz et al., 1994 cited on page 4-122 of the FEIS). Because the Scioto River provides a continual source of new water to the wells, the amount of drawdown in the aquifer is minimized. The effect of additional water withdrawal would also be minimized because the water withdrawal is spread over two well fields, located 6 kilometers apart, and the effect of the additional pumping would be distributed among the two well fields.

4. What is the estimated compressibility of the geologic strata due to increased effective stress from changes in groundwater drawdown?

Staff Response: The geologic strata for the 12 wells in well field X-6609 are described in the Well Log and Drilling Reports for County Permits 571217 through 571228, inclusive. As discussed in response to no. 1 above, the soils are alluvial deposits of the Scioto River and consist almost entirely of granular materials, predominantly fine sand, medium sand, and gravel. The Well Log and Drilling Reports do not include site-specific compressibility data from field or laboratory tests, so typical values for similar soils were used to estimate the soil properties.

For medium dense sands and gravels, typical values for total unit weight range from 119 to 131 pounds per cubic feet (pcf) and for dry unit weight range from 94 to 112 pcf.[1] To estimate the in situ stresses, midrange values of 125 pcf and 103 pcf were used for the total unit weight and dry unit weight, respectively.

The Well Log and Drilling Reports describe the pumping tests performed on each of the 12 wells and present data on pumping rates, well diameter, well depth, screened interval, and initial static water level. The average depth to the groundwater table before pumping was 16.2 feet. The average drawdown, presumably measured within each well, equaled 7.4 feet

The distance from each well to the Scioto River varies between 112 feet and 312 feet. The river acts as a line recharge source and affects the drawdown pattern. Using the reported pumping rates and estimated distances between the wells and the river, the hydraulic conductivity of the soils is estimated to be between 165 to 978 ft/day, which are within the typical range for clean sand and sand and gravel.[2]

The minimum distance from each well to its closest neighbor ranges from 192 feet to 411 feet and is less than the radius of influence for the wells. The estimated additional drawdown at each well due to pumping of an adjacent well is approximately 1 foot. Therefore, an average drawdown of 8.4 feet should be used at the wells to estimate the stress conditions during simultaneous pumping of the wells.

The table below presents an estimate of the in situ vertical stresses with no pumping of the wells and during pumping of the wells. The analysis assumes that the 48-hour pumping tests reached steady-state conditions. The calculations also include several conservative

---

[1] Michael Carter and Stephen P. Bentley, *Correlations of Soil Properties*, at 40 (1991).

[2] U.S. Department of Interior, Water and Power Resources Division, *Ground Water Manual*, at 29 (1977).

assumptions to determine a reasonable upper bound for the maximum subsidence, including the following: The alluvial soils were in a state of virgin compression prior to pumping.

- The alluvial soils were relatively loose.
- The average drawdown at the wells is applied as a uniform drawdown to estimate one dimensional compression down to a depth of 75 feet.

		Initial Conditions - No Drawdown				Uniform Drawdown of 8.4 ft			
Description	Depth (ft)	Total Unit Wt (pcf)*	Total Vertical Stress (psf)	Pore Pressure (psf)	Effective Vertical Stress (psf)	Total Unit Wt (pcf)*	Total Vertical Stress (psf)	Pore Pressure (psf)	Effective Vertical Stress (psf)
Ground Surface	0	--	0	0	0	--	0	0	0
Static Water Depth	16.2	103	1,669	0	1,669	103	1,669	0	1,669
Drawdown Water Depth	24.6	125	2,719	524	2,194	103	2,534	0	2,534
50 ft below ground surface (bgs)	50.0	125	5,894	2,109	3,784	125	5,709	1,585	4,124
75 ft bgs	75.0	125	9,019	3,669	5,349	125	8,834	3,145	5,689

* *The reported values for total unit wt are for depth intervals. For example, under the initial conditions, 103 pcf is the total unit wt for the interval between 0 ft and 16.2 ft. Likewise, under both the initial conditions and uniform drawdown, the 125 pcf in the bottom row is the total unit wt for the interval between 50.0 and 75.0 ft bgs.*

The table below presents values for secant constrained moduli for granular soils.[3] The alluvial soils in the area of the well field which consist primarily of medium sand and gravels almost certainly have a constrained modulus in excess of 4,000 psi.

[3] T. William Lambe and Robert V. Whitman, *Soil Mechanics*, at 155 (1969).

	Relative Density	Secant Constrained Modulus, Virgin Loading (psi): Change in Principal Stress from 9 to 15 psi	Secant Constrained Modulus, Virgin Loading (psi): Change in Principal Stress from 29 to 74 psi
Uniform Gravel 1 mm < D < 5 mm	0	4,400	8,700
	100	17,000	26,000
Well Graded Sand 0.02mm < D < 1 mm	0	2,000	3,700
	100	7,500	17,600
Uniform Fine Sand 0.07 mm < D < 0.3 mm	0	2,100	5,100
	100	7,400	17,400

The calculated maximum change in effective vertical stress due to drawdown is 339 psf. The total calculated subsidence due to groundwater pumping is less than one-half inch, as shown in the table below. The actual subsidence would be less for the following reasons:

- The drawdown away from the wells would be less than at the wells, so the change in effective vertical stress would be less. At a distance of 50 feet from the well line toward the river, the maximum estimated drawdown is 3 feet. At a distance of 50 feet from the well line away from the river, the maximum estimated drawdown is 3.5 feet.
- The drawdown and subsidence estimates use the pumping rate reported in the pump tests, 500 gpm. This equals 720,000 gpd per well, or a total of 8.64 mgd for all12 wells. This exceeds the projected water requirements for the proposed action.
- Past water use was higher than both the current water use and the projected water use under the proposed action. If past drawdown levels exceed current drawdown levels, the alluvial soils have already undergone some compression. Soils do not fully rebound when pumping rates are reduced, so little compression would occur until past pumping rates are exceeded.

Description	Depth (ft)	Change in Effective Vertical Stress (psf)	Vertical Strain	Subsidence within Increment (in)*	Total Subsidence (in)*
Ground Surface	0	0	0	--	--
Static Water Depth	16.2	0	0	0	0
Drawdown Water Depth	24.6	339	0.00059	0.030	0.030
50 ft bgs	50.0	339	0.00059	0.180	0.209
75 ft bgs	75.0	339	0.00059	0.177	0.386

* *The reported values for subsidence within increment and total subsidence are for depth intervals. For example, the subsidence within increment and total subsidence is 0 in for the interval between 0 ft and 16.2 ft. Likewise, the subsidence within increment and total subsidence is 0.030 in for the interval between 16.2 ft and 24.6 ft, and so on.*

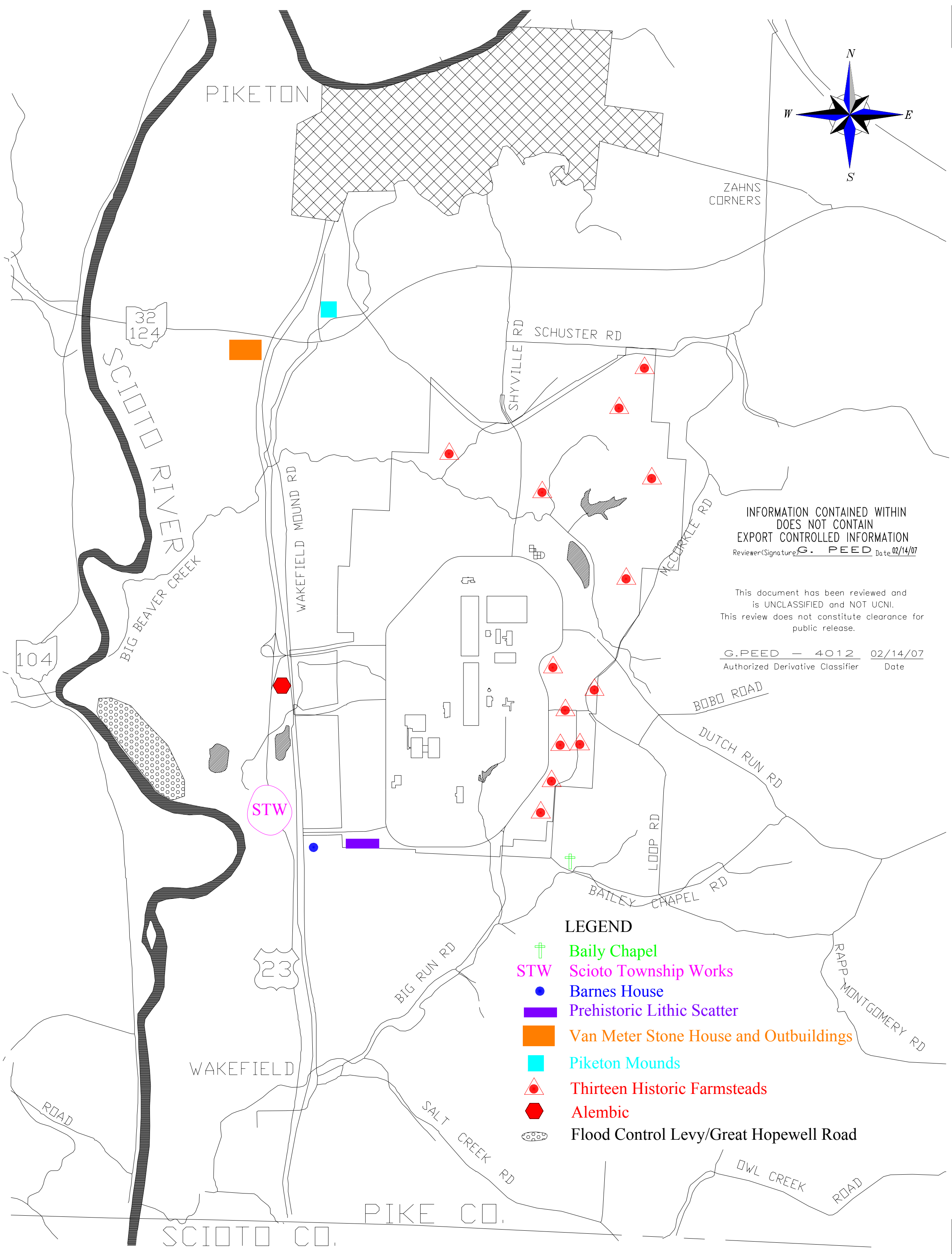
5. What are the estimates of differential settlement at the ground surface from any increase drawdown, and how will it affect the appearance of surface features in the vicinity of the well field?

Staff Response: The magnitude of the reasonable upper bound estimate for total subsidence is less than one-half inch, as discussed above. Differential settlement is defined as the difference between the total settlement measured at two different points, divided by the distance between them. The maximum estimated differential settlement is less than 1/500, a value that would not be visible to the naked eye.

F. Provide maps showing the following (FEIS at 3-9):

1. The location of the following features in relationship to the ACP
   (a) Scioto Township Works
   (b) Piketon Mounds
   (c) Van Meter Stone House and Outbuildings
   (d) Prehistoric lithic scatter
   (e) Thirteen historic farmsteads
   (f) Barnes House
   (g) Bailey Chapel

Staff Response: We have provided a map showing the locations of each of the features listed above. As noted on page 4-5 of the FEIS, none of these sites are located within the area of potential effect for direct effects. The prehistoric lithic scatter and the 13 historic farmsteads, however, are located within the area of potential effect for indirect effects. Three other properties – the Scioto Township Works, the Barnes House, and the Bailey Chapel – are outside the area of potential effect for indirect effects, but close to its boundary. Of all of these sites, the three sites listed on the National Register of Historic Places are the Scioto Township Works, the Piketon Mounds, and the Van Meter Stone House and Outbuildings. All of these features and their locations are described in more detail in Section 3.3 starting on page 3-5 of the FEIS and in Section 4.2.2 starting on page 4-4 of the FEIS.


PIKETON
ZAHNS CORNERS
N
W
E
S
32
124
SCIOTO RIVER
SCHUSTER RD
SHYVILLE RD
WAKEFIELD MOUND RD
BIG BEAVER CREEK
104
MCCORKLE RD
INFORMATION CONTAINED WITHIN
DOES NOT CONTAIN
EXPORT CONTROLLED INFORMATION
Reviewer(Signature) G. PEED Date 02/14/07
This document has been reviewed and
is UNCLASSIFIED and NOT UCNI.
This review does not constitute clearance for
public release.
G.PEED — 4012 02/14/07
Authorized Derivative Classifier Date
BOBO ROAD
DUTCH RUN RD
STW
LOOP RD
BAILEY CHAPEL RD
LEGEND
Baily Chapel
STW Scioto Township Works
Barnes House
Prehistoric Lithic Scatter
Van Meter Stone House and Outbuildings
Piketon Mounds
Thirteen Historic Farmsteads
Alembic
Flood Control Levy/Great Hopewell Road
23
BIG RUN RD
RAPP-MONTGOMERY RD
WAKEFIELD
ROAD
SALT CREEK RD
OWL CREEK ROAD
PIKE CO.
SCIOTO CO.

2. The historic features at each of the sites list above. Do any of these sites relate to the Hopewell Works, Barnes Works, and or the Alembic mentioned in the public comments, and, if so, how? Do any of these relate to the possible section of the Great Hopewell Road illustrated in Appendix B (FEIS at B-229)?

Staff Response: The historic features of each of the sites listed in Question E7.F1 above are shown on the map. Answers to the other two questions included in E7.F2 are provided below.

As indicated on page 3-6 of the FEIS, southern Ohio is best known archaeologically for the Adena and Hopewell Indian mounds (elaborate geometric earthworks, enclosures, and mounds) that were constructed during the Woodland Period (900 B.C. to A.D. 900 (DOE 2004a)). The Piketon Mounds and Scioto Township Works are disturbed remnants of such mounds. Other remnants of Hopewell Indian Mounds are found upstream and downstream of these along the Scioto River, with some of the better preserved or restored examples under the management of the National or state parks systems. Archaeologists have not been able to define interrelationships among the different mound sites with any precision, but it is assumed that some must have been in use during overlapping time periods. The prehistoric lithic scatter 33 Pk 210 likewise may overlap in time of use with some of the mound sites up and down the river, though this has not been demonstrated by archaeologists. As indicated on page 3-11 of the FEIS, "the Barnes Works in Scioto Township" is a reference to the Scioto Township Works, near the Barnes property, upon which the Barnes house is situated. The "Alembic" mentioned in public comments is described as a remnant earthwork that reportedly lies approximately 700 m (2,500 ft) north of the Scioto Township Works. As discussed in Section 4.2.2 starting on page 4-4 of the FEIS, all of these remnant mounds are outside the area of potential effect for direct effects of the project. The Piketon Mound is also located outside the area of potential effect for indirect effects. The prehistoric lithic scatter is located within the area of potential effect for indirect effects and the Scioto Township Works is located outside, but close to the boundary of, the area of potential effect for indirect effects. We believe the potential indirect effects to the prehistoric lithic scatter and the Scioto Township Works would be small, for reasons given on pages 4-5 to 4-7 of the FEIS.

As indicated in the response to comments (public comment number 011-1, FEIS page K-78), local residents informed NRC that the linear feature interpreted as a possible segment of the Great Hopewell Road in public comments was in fact a flood control levy constructed to protect agricultural fields after the devastating floods of 1950. There is no evident connection between the prehistoric earthworks and this relatively recent engineered berm.

G. What is the basis for selecting $25,317 as the average per capita income for the area (FEIS at 3-51)?

Staff Response: This estimate is sourced from:

- (USEC, 2005) USEC Inc. "Environmental Report for the American Centrifuge Plan in Piketon, Ohio." Revision 6. NRC Docket No. 70-7004. November 2005. (ER Table 4.10-3).

The primary source for the estimate is:

- Bureau of Economic Analysis (May 2002). Regional Accounts Data: Local Area Personal Income. <http://www.bea.doc.gov/bea/regional/reis/> (10 April 2003)

Based on the Staff's review of available data, this number is considered appropriate because it is consistent with estimates published by the U.S. Department of Commerce's Bureau of Economic Analysis (BEA) for the four counties in the region of influence of the project. The BEA estimates are based on statistics gathered by the Census Bureau. While there is an inherent and unavoidable degree of uncertainty when forecasting per capita incomes for the years when the project will be active, the Staff verified that the Applicant's approach was economically sound, reasonable, and results in tax revenue estimates that are conservative.

H. What were the models used for the year 2002 to estimate the radiation dose to the maximum exposed individual from the ACP? What input parameters and other assumptions that were made in running the models? (FEIS at 3-64).

Staff Response: The radiation dose values provided in FEIS section 3.13.1 for airborne emissions are originally provided in the DOE Site Environmental Report for the year 2002 ("Portsmouth Annual Environmental Report for 2002." DOE/OR/11-3132 & D1. EQ Midwest, Inc. October 2003). In that report the total radiation dose to the maximum exposed individual from on-site airborne releases is calculated to be 0.031 millirem per year. According to the 2002 Site Environmental Report, 0.026 millirem per year of the 0.031 millirem per year value is generated by airborne releases from on-site facilities managed by USEC, and 0.0046 millirem per year is generated by airborne releases from on-site facilities managed by DOE. USEC and DOE both generate their estimates of radiation dose as part of their NESHAPS compliance demonstration calculations, which are performed using the CAP88 NESHAPS compliance model. User input to the CAP88 model includes descriptions of the radiological characteristics of the release, the physical characteristics of the release source, and the type of food consumption pattern for the affected population. This information is combined with CAP88 input files describing, for the modeling year, the population distribution around the site and the annual average joint frequency weather information. Population data is typically derived from census estimates, and weather information is developed from an on-site meteorological station. Some fundamental assumptions in the modeling include the applicability of the food consumption patterns, as well as assumptions regarding the stability class and dispersion coefficients for the Gaussian Plume model used in the CAP88 model. The US EPA accepts the dose estimates generated by the CAP88 code for showing compliance with the NESHAPS regulatory requirements for radiological releases, so most federal facilities also use the calculations from CAP88 for their site environmental reports.

I. Have the impacts from postulated accidents been evaluated for the ACP, including but not limited to releases during any part of the material flow path from events such as container drops, valve shears, earthquakes, vehicle and aircraft crashes into buildings and storage yards, associated fires, etc. (FEIS at 4-1)?

Staff Response: The Applicant addressed the impacts of accidents such as container drops, valve shears, earthquakes, and vehicle and aircraft crashes into buildings and storage yards in its ISA and ISA Summary. Events are categorized according to the nature of the release mechanism. Some examples of the types of accidents sequences that can occur in the ACP and could exceed the performance requirements of 10 C.F.R. Part 70, Subpart H, are listed in the following table.

Category of Event	**Examples**
Fire	Large fires and small fires in process buildings and cylinder yards
Explosion	Ignition of vehicle fuel or hydrogen
Loss of Containment/Confinement	Cylinder breach, vehicle impact, collapse of cylinders, shearing off valves
Direct Radiological/Chemical Exposure	Direct exposure from spilled hazardous materials
Criticality	Moderator present (water)
External Man Made Hazards	Aircraft crashes, fire in adjacent facility
Natural Phenomena	Earthquakes, flooding

The Applicant documented the estimated consequences of the accident sequences in Appendix D of the ISA Summary. The Staff performed an accident analysis of selected sequences to verify that the consequences of such accidents were determined appropriately and that safety controls were in place for those sequences that could exceed the performance requirements of 10 C.F.R.§ 70.61. The Staff's review is documented in Appendix B of the

SER and in Appendix H of the FEIS which have been withheld from public disclosure because they contain proprietary and/or other sensitive information (e.g., Export Controlled Information). Pages 4-71 to 4-72 of the FEIS include a summary of the selected accident sequences evaluated by the Staff which is publicly available.

Based on the Staff's review of the Applicant's methodology, the selected accident sequences, and the performance of the two site visits, the Staff concludes, with reasonable assurance, that the ISA Summary is complete. The Staff also concludes, based on the aforementioned reasons, that the accident sequences that could exceed the performance requirements have been addressed.

J. The NRC Staff recommends additional mitigation measures to reduce impacts from matter emissions by requiring the use of Tier 2 vehicles and low sulfur diesel fuel. How will this recommendation be documented, required, implemented, and enforced? (FEIS at 4-12).

Staff Response: These additional mitigation measures were recommendations by NRC such that if they were implemented by the Applicant the resulting impacts would be reduced from MODERATE to SMALL. There was no intent that these recommended mitigation measures be made requirements. However, the Applicant did submit a letter dated October 13, 2006, stating their intent to voluntarily implement additional mitigation measures to address this issue. Letter from Steven A. Toelle, USEC, Inc. to Jack R. Strosnider, NRC (Oct. 13, 2006) (ADAMS ML06290466). In particular, the Applicant stated that, to the extent practicable, ACP construction equipment with engine horsepower (HP) ratings of 50 HP or more will utilize Ultra-Low Sulfur Diesel fuel and there will be a stated preference for off-road diesel-powered vehicles and equipment (both mobile and stationary), with engine HP ratings of 50 HP or more, to be Tier 2 compliant.

K. The NRC Staff states that decontamination and decommissioning of old enrichment centrifuges would be controlled by Best Management Practices (BMP) and by utilizing air filtration and trapping systems in order to capture releases. How will these controls be required, implemented, and enforced? (FEIS at 4-12).

Staff Response: The decommissioning of old enrichment centrifuges has already been completed. The work was performed by USEC under contract to the Department of Energy (DOE) and under DOE regulatory authority. DOE staff routinely performed inspections of the activities to ensure compliance with the applicable regulations and requirements.

L. Clarify, based on the NRC Staff's review, that the size of the disturbed area needed for the new cylinder yard does not require detention ponds to reduce the peak runoff to the neighboring streams and to provide control for potential radiological releases. What was the technical justification that demonstrated to the Staff that these ponds are not required as a component of the BMPs for controlling runoff? (FEIS at 4-19).

Staff Response: The Staff reviewed the potential impacts associated with peak storm water runoff during construction of the proposed new cylinder yard, as well as potential radiological releases from the yard during operation. In addition, the Applicant used estimated cross sections and Manning's formula[13] with n = 0.15, a value typical for floodplains and very poor natural channels, to estimate the peak runoffs of the streams around the reservation. The Applicant found that the proposed cylinder storage yard would result in a net increase in peak runoff storm flow of 74 cubic feet per second (CFS) to the neighboring streams during construction. This value indicates that the need for a storm water detention pond would not be a necessary design element.

---

[13] One the most commonly used equations governing Open Channel Flow is known as the Mannings's Equation. It was introduced by the Irish Engineer Robert Manning in 1889 as an alternative to the Chezy Equation. The Mannings equation is an empirical equation that applies to uniform flow in open channels and is a function of the channel velocity, flow area and channel slope. The "n" value refers to Manning's Roughness Coefficient.

As discussed in FEIS Section 4.2.5.1, engineering controls and best management practices including erosion control ditches, temporary vegetation seeding, and silt fencing would be used to control both the peak runoff rates and associated sedimentation by detaining storm water and reducing its velocity. As discussed in FEIS Section 4.2.6.1, the Applicant has agreed to design and construct the proposed cylinder storage yard in a manner that would preserve the existing upland hardwood forest and the riparian forest adjacent to the managed and old field areas where the cylinder storage yard would be constructed. The preservation of the upland and riparian forests would further reduce the peak runoff rates and limit sediment transport during construction.

During operation, storm water runoff from the north pads (X-745G-2 and X-745H) would drain to holding ponds in accordance with a service agreement and would be continuously monitored with automated samplers. The Applicant's modeled net increase in peak runoff values from the north pads, 74 CFS, would not exceed the handling and operating capacity of the existing holding ponds in meeting their discharge requirements In addition, the control of radiological releases from the cylinder storage pads includes the cylinder design and handling practices, as well as the Applicant's proposed inspection and maintenance program for its UF_6 cylinders. This inspection and maintenance program is intended to ensure that no licensed material is released from the storage pads, and monitoring data from the holding ponds would be available to ACP environmental personnel as assurance that no unanticipated discharge occurred.

M. To what degree has the NRC Staff reviewed the impact analysis to ascertain if the potential impacts of increased peak runoff rates on the biota in the adjacent streams is caused by the increased impervious area of the storage yard (FEIS at 4-26)?

Staff Response: The Staff reviewed the potential impacts associated with increased peak runoff rates associated with the construction of the 24-acre cylinder storage yard, as well as the impacts associated with an additional 24 acres of impervious surface. During construction, as discussed in FEIS Section 4.2.5.1, engineering controls and best management practices including erosion control ditches, temporary vegetation seeding, and silt fencing would be used to control both the peak runoff rates and associated sedimentation that would affect the biota by detaining storm water and reducing its velocity. As discussed in FEIS Section 4.2.6.1, USEC has agreed to design and construct the proposed cylinder storage yard in a manner that would preserve the existing upland hardwood forest and the riparian forest adjacent to the managed and old field areas where the cylinder storage yard would exist. The preservation of the upland and riparian forests would further reduce the peak runoff rates and limit sediment transport during construction.

Once the cylinder storage yard had been constructed, storm water from the storage yard would be collected and transported to existing holding ponds. The Applicant's modeled net increase in peak runoff storm flow value from the north pads, 74 cubic feet per second (CFS), would not exceed the handling and operating capacity of the existing holding ponds in meeting their discharge requirements. Therefore, the peak runoff rates would be processed through existing infrastructure within their required discharge requirements and would not result in a change in the impacts on the biota in the receiving streams.

N. Is the 76 centimeter dimension for the heeled cylinders a length, a radius or a diameter? Explain the process which generates these heeled cylinders, including the composition of the material (e.g., DUF_6, EUF_6, both, or other). (FEIS at 4-79).

Staff Response: A gas centrifuge enrichment process is basically a mechanical process that takes natural uranium [in the form of uranium hexafluoride (UF_6)] and separates it into a product stream enriched to about 5 percent U^{235} and a tails stream depleted to about 0.3-0.4 percent of the U^{235} isotope. All cylinders involved in the transportation and storage of UF_6, whether natural, enriched or depleted, must meet the requirements of ANSI N14.1, "American National Standard for Nuclear Materials - Uranium Hexafluoride - Packaging for Transportation."

The natural uranium feed material, in the form of UF_6, is shipped to the facility in 10-ton Mark 48X or 14-ton Mark 48Y cylinders. The UF_6 is removed from the cylinders and fed into the cascade through a sublimation process facilitated by high vacuum conditions. The cylinder will remain hooked up to the process until almost all of the material has been removed.
The remaining material that cannot be practically removed is referred to as the "heels." A heel is a residual amount of UF_6 and nonvolatile reaction products of uranium (see ANSI N14.1, page 9). A cylinder is technically considered "empty" when the heel is less than the quantity specified in ANSI N14.1, Section 8.1.2 (50 pounds for any Mark 48 type). Note that this definition should not be confused with the category of empty packaging used in 49 C.F.R. § 173.427. Once "emptied", the feed cylinders can be returned to the conversion facility for reuse.

As the UF_6 is processed in the cascade, product (enriched UF_6 or EUF_6) is drawn out into a Mark 30B 2.5-ton cylinder, which is 30 inches (about 76 centimeters) in diameter. This product is shipped to one of several low enriched uranium fuel fabrication facilities. The "empty" product cylinders will be returned to the ACP for reuse. Per ANSI N14-1, an empty Mark 30B cylinder is limited to a heel of less than or equal to 25 pounds. For Type 30B product cylinders, the heel material is expected to be EUF_6. The tails, or DUF_6 stream, will be feed into 14-ton Mark 48 G cylinders.

Cylinder dimensions are specified in ANSI N14.1. Selected examples follow:

Type of Cylinder	Outer Diameter	Length
30B	30±1/4 in.	76±½ in.
48X	48-1/2±1/4 in.	86-1/2±3/8 in.
48Y	48-1/2±1/4 in.	117-1/2±3/8 in.
48G	48-5/8+1/-½ in.	116-7/8±3/8 in.

O. Does the analysis of the cumulative impact of the proposed action include decommissioning as well as construction and operation, listed in the first bullet item of FEIS page 4-100?

Staff Response: Because decontamination and decommissioning activities for the proposed ACP are anticipated to occur approximately 30 years in the future, only a general description of the activities that would be conducted for the proposed ACP can be developed at this time for the FEIS. In accordance with 10 C.F.R. § 70.38(d) and 10 C.F.R. § 70.38(g)(1), the licensee would be required to prepare and submit a Decommissioning Plan to the NRC at least twelve months prior to the expiration of the NRC license, and would begin the decontamination and decommissioning activities upon approval of the final Decommissioning Plan by the NRC. Under 10 C.F.R. § 70.38(g)(4), the Decommissioning Plan would include a description of the planned decommissioning activities, including: site characterization information and site remediation plan; a description of the methods used to ensure protection of workers and the environment against radiation hazards during decommissioning; a description of the planned final site radiation survey; an updated detailed cost estimate for the activities; and a description of the physical security plan and the material control and accounting plan for the decommissioning. The Decommissioning Plan would be subject to NEPA review (including an evaluation of cumulative impacts), as applicable, at the time the Plan is submitted to the NRC.

P. What are the aquifer parameters that were used to derive the numbers needed to calculate that the withdrawal rate would be 31% of the system capacity (FEIS at 4-108)?

Staff Response: The Staff did not use any aquifer parameters to calculate the 31%, but rather used the current and proposed water withdrawal rates presented in Table 4-25. Specifically, the 31% was calculated as follows:

- The current groundwater withdrawal rate is 5,500,000 gpd;
- The ACP would increase withdrawals by 650,000 gpd and the DUF_6 conversion facility would increase withdrawals by 90,411 gpd, for a total increase of 740,411 gpd;
- The combined withdrawal rate would therefore be 6,240,411 gpd (5,500,000 plus 740,411); and
- 6,240,411 gpd is approximately 31% of the current 20,000,000 gpd system design capacity reported by USEC.

CONCLUSION

The above, along with Appendix A, constitute the Staff's response to the Board's February 6 Order.

Respectfully submitted,

/RA by Margaret J. Bupp/

Margaret J. Bupp
Brett M. Klukan
Counsel for the NRC Staff

Dated at Rockville, Maryland
this 20th day of February, 2007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
USEC Inc.	)	Docket No. 70-7004
	)	
(American Centrifuge Plant)	)	ASLBP No. 05-838-01-ML
	)	

NOTICE OF APPEARANCE

Notice is hereby given that the undersigned attorney herewith enters an appearance in the above-captioned matter. In accordance with 10 C.F.R. § 2.314(b), the following information is provided:

Name: Brett M. Klukan

Address: U. S. Nuclear Regulatory Commission
Office of the General Counsel
Mail Stop: O-15 D21
Washington, D.C. 20555

Telephone Number: 301-415-3629

E-mail Address: bmk1@nrc.gov

Facsimile: 301-415-3725

Admissions: State of Pennsylvania

Name of Party: NRC Staff

Respectfully submitted,

/RA by Brett M. Klukan/

Brett M. Klukan
Counsel for the NRC Staff

Dated at Rockville, Maryland
this 20th day of January, 2007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
USEC, Inc.	)	Docket No. 70-7004
	)	
(American Centrifuge Plant)	)	ASLBP No. 05-838-01-ML

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE TO ATOMIC SAFETY AND LICENSING BOARD ORDER OF FEBRUARY 6, 2007" and "NOTICE OF APPEARANCE" for Brett M. Klukan in the above captioned proceeding have been served on the following persons by deposit in the United States Mail; through deposit in the Nuclear Regulatory Commission internal mail system as indicated by an asterisk(*); and by electronic mail as indicated by a double asterisk (**) on this 20th day of February, 2007.

Administrative Judge * **
Lawrence G. McDade, Chair
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Mail Stop: T-3F23
Washington, D.C. 20555
E-Mail: lgm1@nrc.gov

Administrative Judge * **
Peter S. Lam
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Mail Stop: T-3F23
Washington, D.C. 20555
E-Mail: psl@nrc.gov

Atomic Safety and Licensing Board Panel *
U.S. Nuclear Regulatory Commission
Mail Stop: T-3F23
Washington, D.C. 20555

Office of the Secretary * **
Attn: Rulemaking and Adjudications Staff
U.S. Nuclear Regulatory Commission
Mail Stop: O-16 C1
Washington, D.C. 20555
E-mail: hearingdocket@nrc.gov

Administrative Judge * **
Dr. Richard E. Wardwell
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Mail Stop: T-3F23
Washington, D.C. 20555
E-Mail: rew@nrc.gov

Office of Commission Appellate Adjudication *
U.S. Nuclear Regulatory Commission
Mail Stop: O-16 C1
Washington, D.C. 20555

Dennis J. Scott, Esq. **
USEC Inc.
6903 Rockledge Drive
Bethesda, MD 20817
E-mail: scottd@usec.com

Donald J. Silverman**
Alvin H. Gutterman**
Martin O'Neill**
Morgan Lewis & Bockius, LLP
1111 Pennsylvania Ave., N.W.
Washington, D.C. 20004
E-mail: dsilverman@morganlewis.com
agutterman@morganlewis.com
martin.o'neill@morganlewis.com

Debra Wolf
Law Clerk
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Mail Stop: T3 F23
Washington, D.C. 20555-0001
E-Mail: daw1@nrc.gov

/RA by Margaret J. Bupp/

Margaret J. Bupp
Counsel for the NRC Staff