June 1, 2006

Mr. Russell B. Starkey, Jr.
Vice President - Operations
United States Enrichment Corporation
Two Democracy Center
6903 Rockledge Drive
Bethesda, MD 20817

SUBJECT: INSPECTION REPORT NO. 70-7001/2006-201**(CORRECTED)**

Dear Mr. Starkey:

This letter transmits a revised version of the original inspection report. The U.S. Nuclear Regulatory Commission (NRC) conducted a routine, scheduled, and announced criticality safety inspection from May 1 - 5, 2006, at the Paducah Gaseous Diffusion facility in Paducah, Kentucky. The purpose of this inspection was to determine whether activities authorized by your certificate involving special nuclear material were conducted safely and in accordance with regulatory requirements. Throughout the inspection, observations were discussed with your staff. An exit meeting was held on May 5, 2006, during which time inspection observations and findings were discussed with your management and staff.

The inspection, which is described in the enclosure, focused on: (1) the most hazardous activities and plant conditions; (2) the most important controls relied on for safety and their analytical basis; and, (3) the principal management measures for ensuring controls are capable, available, and reliable to perform their functions relied on for safety. The inspection consisted of analytical basis review, selective review of related procedures and records, examinations of relevant NCS-related equipment, interviews with NCS engineers and plant personnel, and facility walkdowns to observe plant conditions and activities related to safety basis assumptions and related NCS controls. Based on the inspection, your activities involving nuclear criticality hazards were found to be conducted safely and in accordance with regulatory requirements.

In accordance with 10 CFR 2.390 of NRC's "Rules of Practice," a copy of this letter and the enclosure will be available in the public electronic reading room of the NRC's Agency-Wide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm/adams.html.

If you have any questions concerning this report, please contact Christopher Tripp, of my staff, at (301) 415-6215.

Sincerely,

/RA/

Melanie A. Galloway, Chief
Technical Support Section
Special Projects Branch
Division of Fuel Cycle Safety
and Safeguards

Docket No.: 70-7001

Enclosure: Inspection Report No. 70-7001/2006-201

cc: S. Penrod, Paducah General Manager
S. R. Cowne, Paducah Regulatory Affairs Manager
P. D. Musser, Portsmouth General Manager
S. A. Toelle, Director, Nuclear Regulatory Affairs, USEC
R. M. DeVault, Regulatory Oversight Manager, DOE
G. A. Bazzell, Paducah Facility Representative, DOE
Janice H. Jasper, State Liaison Officer

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U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Docket Number:	70-7001
Certificate Number:	GDP-01
Report Number:	70-7001/2006-201
Certificatee:	United States Enrichment Corporation
Location:	Paducah, Kentucky
Inspection Dates:	May 1 - 5, 2006
Inspectors:	Christopher Tripp, Sr. Criticality Safety Reviewer Natreon Jordan, Criticality Safety Inspector
Approved by:	Melanie A. Galloway,Chief Technical Support Section Special Projects Branch Division of Fuel Cycle Safety and Safeguards

Enclosure

United States Enrichment Corporation
Paducah Gaseous Diffusion Plant

NRC Inspection Report
70-7001/2006-201

EXECUTIVE SUMMARY

Introduction

Staff of the U. S. Nuclear Regulatory Commission (NRC) performed a routine, scheduled, and announced criticality safety inspection of the Paducah Gaseous Diffusion Plant (GDP) in Paducah, Kentucky, from May 1 - 5, 2006. The inspection included an on-site review of certificatee programs dealing with plant operations, the nuclear criticality safety (NCS) program, audits and inspections, and NCS-related corrective actions. The certificatee programs were acceptably directed toward the protection of public health and safety and were in compliance with NRC regulatory requirements. The inspection focused on risk-significant fissile material processing activities including Buildings C-310, C-335, C-337, C-360, C-400, C-746-Q1, and C-754.

Results

- A concern was identified regarding inclusion of mixed oxide- and high-enriched uranium-(HEU) driven benchmark experiments in the validation report.

- A concern was identified regarding a nuclear criticality safety analysis failing to contain the basis for approved storage of containers with potentially better moderators than water in arrays of containers of fissile material.

- All other nuclear criticality safety analyses and supporting calculations reviewed demonstrated adequate identification and control of NCS hazards to assure operations within subcritical limits.

- Observed plant operations were conducted safely and in accordance with regulatory requirements.

REPORT DETAILS

1.0 NCS Program (88015)

a. Scope

The inspectors reviewed NCS analyses to determine that criticality safety of risk-significant operations was assured through engineered and human performance (controls) with adequate safety margin/certainty and preparation and review by qualified staff. The inspectors reviewed selected aspects of the following documents:

- NCSE [Nuclear Criticality Safety Evaluation]-063, “Liquid Uranium Salvage Operations in the C-710 Laboratory Facility,” Revision 4, dated October 20, 2005
- NCSE-085, “Operation of the C-400 Cylinder Washing, Hydrostatic Testing, and Drying Facility,” Revision 4, dated September 1, 2005
- NCS-RG-05-001, “NCS Remediation Guide,” Revision 4
- DAC-832-ZA-1280-0042, “Calculations for 5.5-Gallon Drums, 2.1-Gallon Drums, and 21-Liter Carboy Containers,” Revision 0, dated April 10, 2006
- NCSE-091, “Fissile/Potentially Fissile Waste Container Storage and Handling,” Revision 5, dated April 13, 2006
- CP2-EW-WM1036, “Nuclear Criticality Safety Implementation Requirements for Handling and Storage of Fissile and Potentially Fissile Waste,” dated January 31, 2006
- KY/S-221, “Validation of the SCALE 4.4 Nuclear Criticality Safety Code System and the ENDF/B-IV 27-Group Cross-Section Library at the Paducah Gaseous Diffusion Plant,” Revision 6, dated January 2005
- KY/G-748, “Validation of the MCNP-5 Nuclear Criticality Safety Code System Using the ENDF/B-V Cross-Section Library at the Paducah Gaseous Diffusion Plant,” Revision 01, dated February 2005
- NCSE-036, “C-400 Alkali Tank,” Revision 3, dated March 6, 2006
- NCSE-046, “Operation of the ICP Laboratory in C-3170,” Revision 00
- NCSE-063, “C-710 Liquid Uranium Salvage Operations,” Revision 4, dated February 10, 2006
- NCSE-074, “Normetex Pump Recycle Line and Vapor Phase Transfer,” Revision 1, dated April 17, 2006
- DAC-832-ZA1280-0011, “Handling of Carboys in Liquid Uranium Salvage Operations in C-710," Revision 0
- DAC-832-ZA1280-0027, “Handling and Storage of Carboys in Liquid Uranium Salvage Operations in Room 15 in C-710,” Revision 0
- NCSE-097, “C-400 Uranium Recovery System at the Paducah Gaseous Diffusion Plant,” Revision 00, dated March 8, 2004
- DAC-832-ZA1280-0049, “C-400 Uranium Recovery System–KENO Calculations”

b. Observations and Findings

The inspectors reviewed the SCALE 4.4 and MCNP-5 validation reports to determine whether they established an acceptable upper safety limit for plant operations. The inspectors determined that the statistical methodology was applied correctly and that the chosen critical experiments were consistent with the derived area of applicability (AOA). The inspectors noted that Section 5.2 of the Safety Analysis Report (SAR) commits the certificatee to a 95/99.9 lower tolerance band approach, whereas the two validation reports calculated upper safety limits based on the 95% one-sided confidence band. The certificatee did a comparison to determine which method yielded the lowest result, and used that as its k_{eff} limit. In both cases, the upper safety limit was greater than the Technical Safety Requirement (TSR) limit of 0.9634. The inspectors conclude that the use of the TSR limit as the criterion for subcriticality on the documents reviewed is appropriate.

During their review of the validation reports, the inspectors noted that several mixed-oxide (MOX)- and HEU-driven experiments were included in the validation database. The inspectors questioned the certificatee about this, because Inspection Report 70-7001/2005-201 had closed out a similar issue. In that Inspection Report, Inspection Followup Item (IFI) 70-7001/2004-203-01 had been closed with the statement that "the certificatee had completed reanalysis of the MOX benchmarks and revised the validation report...to remove the MOX benchmarks." The inspectors determined that these MOX benchmarks had not actually been removed; the certificatee had merely recalculated the code bias with and without the MOX benchmarks and concluded that the effect on the bias was negligible. While the effect on the bias was negligible, the inspectors questioned whether the inclusion of these benchmarks had an effect on the validated AOA.

The certificatee stated that it had included the MOX-driven benchmarks to validate the low enrichment range of the AOA. These six benchmarks all involved natural uranium (with an enrichment of 0.711wt% ^{235}U). The inspectors determined that without these MOX-driven benchmarks, the enrichment range would end at 1.4wt% ^{235}U. Below 1wt% ^{235}U, the pure uranium systems at the Paducah GDP cannot achieve criticality. Thus, the MOX-driven benchmarks are needed to validate calculations in the range of 1-1.4 wt% ^{235}U. These benchmarks are also needed for inclusion of zircalloy, lead, and boron in the AOA. The HEU-driven benchmarks (which also contain uranium enriched to 4.46 wt% ^{235}U) were included to validate calculations at the low end of the range in hydrogen-to-^{235}U ratio (H/X) and calculations involving organic material, aluminum, and stainless steel (SS304).

The inspectors determined that there were other experiments containing aluminum and SS304, as well as a limited number of other experiments at the lower end of the range in H/X. "Organic material" included any organic tape, plastic, glue, vinyl, and hydrocarbon materials (e.g., paraffin and hydrocarbon oil) that might occur in the benchmark cases. The inspectors determined that inclusion of the HEU-driven experiments did not appear to significantly impact the AOA. However, this was not the case with the MOX-driven experiments. The certificatee stated that operations were not limited in enrichment to the 1-1.4 wt% ^{235}U range, and that this range needed to be validated mainly to support parametric studies. As an example, the certificatee provided the safe mass curve included in TSR 2.5, Appendix B. The inspectors determined that, at enrichments below 1.4wt% ^{235}U, the minimum mass required for criticality (assuming optimal moderation, geometry, and reflection conditions) exceeded 700 lb U, and went up to 10,000 lb U at 1wt% ^{235}U. The inspectors therefore determined that the lack of data in the 1-1.4wt% ^{235}U range did not appear to be very safety significant. However, the certificatee did not have an analysis showing what the effect of excluding the MOX-driven benchmarks would be on the other aspects of the AOA. Determining the effect of inclusion of the MOX-driven benchmarks on the AOA and justifying their inclusion will be tracked as **Inspection Followup Item (IFI) 70-7001/2006-201-01.**

The inspectors also reviewed several recently revised NCSEs to determine whether calculations were performed within the scope of the validation reports. The specific NCSEs reviewed were NCSE-036, NCSE-063, and NCSE-074. The inspectors also reviewed several supporting design analysis calculations (DACs) as listed above. In each case, the inspectors determined that the calculations were appropriately within the scope of the validated AOA. However, the level of demonstration varied widely between the various analyses, even though they were all of recent vintage. In some cases, the inspectors had to calculate the H/X ratio from the material description to confirm that it was within the validated AOA, because this information was not included in the analysis. In addition, in all the cases reviewed, there did not appear to be any discussion of the average energy group (AEG). The certificatee stated that it did routinely consider the AEG in determining whether calculations were within the validated AOA, but that this was not always clear from the documentation. In the case of NCSE-046, the certificatee showed the inspectors an analysis justifying the modeling of hexane and triethyl hexaphosphate (TEHP) based on a calculation of hydrogen number density compared to that of water. Similarly, while benchmarks did not specifically contain hydrocarbon oils, the atom density of included materials (e.g., paraffin, polyethylene) appeared to appropriately provide for the use of similar organic materials.

The certificatee committed to develop a consistent minimum level of documentation for verifying the compliance of future facility calculations with the validated AOA. The development of this minimum level of documentation will also be tracked as part of **IFI 70-7001/2006-201-01**. The inspectors also reviewed the two most recent semi-annual verification reports and determined that they were done appropriately.

Within the selected aspects reviewed, the inspectors determined that the analyses were performed by qualified NCS engineers, that independent reviews of the evaluations were completed by qualified NCS engineers, that subcriticality of the systems and operations was assured through appropriate limits on controlled parameters, and that double contingency was assured for each credible accident sequence leading to inadvertent criticality. The inspectors determined that NCS controls for equipment and processes assured the safety of the operations.

c. Conclusions

With the exception of IFI 70-7001/2006-201-01, code validation and verification was done appropriately. Calculations were done within the scope of the validated AOA, though there was considerable variation in demonstrating AOA compliance. Reviewed nuclear criticality safety analyses and supporting calculations demonstrated adequate identification and control of NCS hazards to assure operations within subcritical limits. The NCS program as observed was adequate for maintaining acceptable levels of safety.

2.0 NCS Inspections, Audits and Investigations (88015)

a. Scope

The inspectors observed the certificatee's responses to a non-reportable event and an occurrence identified by NRC inspectors during the inspection to ensure that the certificatee could adequately identify the root cause and implement appropriate corrective actions to prevent similar occurrences.

b. Observations and Findings

While performing a walkdown of the C-337 facility with NCS staff, inspectors identified a number of leaks on the floor. Most of these leaks were dried, but one of them, resulting in a large pool of water on the cell floor, appeared to be the result of a slow leak from a valve tagged with a "material deficiency" tag. Several of these leaks were yellowish in appearance and consisted of several concentric rings, while others consisted of dried splatters that were coming from corroding flange connections in the overhead piping structure. The certificatee stated that the stains were calcium phosphate deposits, but these should have been whitish rather than yellowish in color. The certificatee then stated that there were several anti-corrosive and other additives to the recirculating cooling water (RCW) that could have resulted in the yellowish tinge. At the inspectors' request, a health physics survey was conducted and found no indication of uranium in the RCW water. Through walkdowns, health physics sampling, and discussions, the certificatee was able to conclude that the leaks, although a result of deteriorating piping in the flange areas, contained an inert mixture of water and various anti-corrosive agents. The certificatee stated that those piping areas are included in its corrective action program and are scheduled to be repaired.

The inspectors walked down the overhead piping and concluded that it was part of the RCW system. The certificatee stated that the condition of RCW was not an NCS issue, but a material degradation issue that is covered by its System Health Program. The inspectors reviewed NCSE-095, "Operation and Shutdown of the Diffusion Cascade," Rev. 02, dated April 14, 2005, and determined that the integrity of the RCW barrier was not credited as one of the safety-related items (SRIs) in NCSE-095. This is because the main coolant loop between the RCW system and the process gas is maintained at a higher pressure, such that it would require both a pressure drop and a loss of integrity of both secondary and primary coolant to allow moderation of the process gas. However, the inspectors were concerned about the integrity of the RCW system due to the fact that the suspect piping structure provided a heat sink to very important lines in the process. Significant degradation of the coolant system integrity could cause an over-temperature situation within the cascade.

Another event took place during the inspection regarding radioactive waste being improperly stored in a facility. The certificatee found, during a walk-down, a plastic bag containing a piece of flexible tubing stored in an array with containers of uncharacterized material. Containers of uncharacterized material are stored in an array with a passive engineered control which maintains spacing between containers. This control consists of a metal pan limiting spilled material to less than a safe slab depth, with a central slot for holding 5.5-gallon drums containing uncharacterized material. Other materials are not approved for storage in these pans because they are outside the scope of the analysis demonstrating subcriticality for this system. The certificatee found the bag containing the tube in one of the slot areas while in the array. The certificatee was investigating this event during the inspection to determine the root cause. The inspectors concluded that the event was of low risk significance because no moderation was present and spacing between the pan and other potentially fissile material was maintained.

c. Conclusions

The certificatee's response to operational events during the inspection was appropriate.

3.0 Plant Operations (88015)

a. Scope

The inspectors performed plant walkdowns to review activities in progress and to determine whether risk-significant fissile material operations were being conducted safely and in accordance with regulatory requirements. The inspectors verified the adequacy of management measures for assuring the continued availability, reliability, and capability of safety-significant controls relied upon by the certificatee for controlling criticality risks to acceptable levels. The inspectors performed walkdowns of Buildings C-310, C-335, C-337, C-360, C-400, C-746-Q1, and C-754.

The inspector reviewed selected aspects of the following documents prior to performing the walkdowns:

- NCSE-063, "Liquid Uranium Salvage Operations in the C-710 Laboratory Facility," Revision 4, dated October 20, 2005
- NCSE-085, "Operation of the C-400 Cylinder Washing, Hydrostatic Testing, and Drying Facility," Revision 4, dated September 1, 2005
- NCS-RG-05-001, "NCS Remediation Guide," Revision 4
- DAC-832-ZA-1280-0042, "Calculations for 5.5-Gallon Drums, 2.1-Gallon Drums, and 21-Liter Carboy Containers," dated April 10, 2006
- NCSE-091, "Fissile/Potentially Fissile Waste Container Storage and Handling," Revision 5, dated April 13, 2006
- CP2-EW-WM1036, "Nuclear Criticality Safety Implementation Requirements for Handling and Storage of Fissile and Potentially Fissile Waste," dated January 31, 2006
- NCSE-108, "Operation of the Fixed High Efficiency Filter Systems in C-310 and C-360," Revision 00, dated February 10, 2006
- CP2-CO-CO1034, "Out-of-Service/Abandoned-in-Place/Spare Equipment Control," Revision 9, dated October 27, 2005

b. Observations and Findings

The inspectors verified that controls identified in the NCS analyses reviewed were installed or implemented and were adequate to assure safety. The cognizant NCS engineers were knowledgeable and able to explain the basis for changes in operations and controls.

While on a walkthrough of the C-754 Building, inspectors identified drums of spacing-exempt NCS material being stored in an array with large drums containing potentially better moderating and/or reflecting material than water. The certificatee used water to analyze normal and credible accident scenarios involving storage configurations in that area. Drums containing spacing-exempt NCS material are allowed to be stored next to other spacing-exempt NCS material and drums verified not to contain NCS material. During a walkthrough, inspectors identified an array of spacing-exempt material stored in an array of 55-gallon drums containing oil. Oil can be shown to provide better interstitial moderation than water in some cases. The analysis governing this storage area (NCSE-091, "Fissile/Potentially Fissile Waste Container Storage and Handling") looked at storage configurations of drums containing these materials. However, justification for storage of drums with spacing-exempt NCS material in arrays with drums of potentially better moderators was not explicitly identified in the analysis. The certificatee was able to provide justification for the storage array with scenarios in the analysis in combination with a corresponding analysis (DAC-832-ZA-1280-0042, "Calculations for 5.5-Gallon Drums, 2.1-Gallon Drums, and 21-Liter Carboy Containers"). The modeled scenarios included optimum moderation, so that any additional moderation would decrease the reactivity of the system. Inspectors communicated to the certificatee that an acceptable justification for approved storage configurations needed to be documented in the corresponding NCS analysis.

During the inspection, the certificatee committed to revise the NCSE-091 analysis to include justification for storage of drums of spacing-exempt NCS material next to drums containing potentially better moderators than water. The certificatee's actions to revise the NCSE-091 analysis will be tracked as **Inspection Followup Item (IFI) 70-7001/2006-201-02**.

In addition, the inspectors walked down a number of other drum storage areas, noting that the drums containing uncharacterized material were very similar in nature to those that were spacing exempt. While labeling provided some basis for distinguishing these two types of drums, the drums were plastered with many labels that could not be readily distinguished at a distance. Also, there were no postings at the periphery of spacing-exempt arrays indicating that uncharacterized drums could not be stored in that area. The inspectors noted that the lack of postings appeared to create the potential for violation of spacing requirements.

The inspectors also walked down the process ventilation system in the C-310 and C-360 Buildings and observed two examples of out-of-service equipment in the C-360 Building. With regard to process ventilation, the inspectors reviewed NCSE-108 to determine whether NCS-related SRIs were consistent with the configuration of the ventilation equipment in the field, and with drawings. The inspectors determined that the ventilation system met all NCS requirements and was as described in facility drawings. With regard to the out-of-service equipment, the inspectors examined a bank of technetium traps and a cold trap in the basement of the C-360 Building. The inspectors observed that valves on the technetium traps were tagged with paper “out-of-service” tags, but the equipment was not physically disconnected from fissile bearing processes. When questioned about this, the certificatee stated that the equipment was considered out of service, but not abandoned in place. While the technetium traps had not been used for some time, there was no intent to remove them from the facility’s configuration control program or delete NCS controls applied to them. By contrast, there was a blue sign leaning against the cold trap that indicated that it was abandoned in place. Inspectors confirmed from plant drawings (M5E-144430-A1 Rev. 5; J5E-14443-2006 Rev. 4; P5E-14443-3005 Rev. 19; and P5E-14443-3007, Rev. 14) that the cold trap had been physically disconnected from any fissile material operations by cutting and capping the transfer piping. In addition, NCSE-059 specifically stated that the cold trap had been abandoned in place and had specific requirements (including a 2 foot edge-to-edge spacing and isolation caps on the transfer piping) to maintain NCS. Inspectors also reviewed procedure CP2-CO-C01034 and determined that the requirements for abandoned-in-place equipment were being followed.

c. Conclusions

Observed plant operations were conducted safely and in accordance with regulatory requirements. The analytical basis for storing drums of oil with spacing-exempt storage arrays was not appropriately documented and resulted in an IFI. Labeling and posting practices could create possible confusion that could lead to a violation of NCS controls. Procedures for out-of-service and abandoned-in-place equipment appeared adequate, and plant NCS documents had been appropriately revised to provide for NCS for this equipment.

4.0 Exit Meeting

The inspectors communicated the inspection scope and results to members of Paducah Gaseous Diffusion Plant management and staff throughout the inspection and during an exit meeting on May 5, 2006. Paducah Gaseous Diffusion Plant management and staff acknowledged and understood the findings as presented.

Supplementary Information

1.0 List of Items Opened, Closed, and Discussed

Opened

IFI 70-7001/2006-201-01	Tracks the certificatee's justification for inclusion of MOX-driven benchmarks in the validation reports, and establishment of criteria for demonstration of AOA compliance in facility calculations.
IFI 70-7001/2006-201-01	Tracks the certificatee's development of guidelines which will ensure a consistent minimum level of documentation for verifying the compliance of future facility calculations with the validated AOA
IFI 70-7001/2006-201-02	Tracks the certificatee's revision of NCSE-091 to include justification for storage of drums of spacing-exempt NCS material next to drums containing potentially better moderators than water.

Discussed

None.

Closed

None.

2.0 Inspection Procedures Used

IP 88015	Headquarters Nuclear Criticality Safety Program

3.0 Partial List of Persons Contacted

USEC

T. Henson	Manager, Nuclear Criticality Safety
S. Cowne	Manager, Nuclear Regulatory Affairs
D. Stadler	Engineer, Nuclear Regulatory Affairs
T. Hofer	Engineer, Nuclear Criticality Safety
J. Lewis	Manager, Maintenance
D. Baltimore	Engineer, Nuclear Criticality Safety
B. Chenier	Nuclear Criticality Safety
M. Boren	Nuclear Regulatory Affairs
S. Penrod	General Manager
E. Paine	Manager, Chemical Operations
L. Jackson	Manager, Operations

NRC

C. Tripp	Sr. Criticality Safety Reviewer, Headquarters
N. Jordan	Criticality Safety Inspector, Headquarters
M. Thomas	Resident Inspector, RII
N. Rivera	Inspector, RII

All attended the exit meeting on May 5, 2006.

4.0 List of Acronyms and Abbreviations

ADAMS	Agency-Wide Document Access and Management System
AEG	average energy group
AOA	area of applicability
CFR	Code of Federal Regulation
DAC	design analysis calculation
DOE	U.S. Department of Energy
HEU	high-enriched uranium
H/X	hydrogen to ^{235}U ratio
GDP	gaseous diffusion plant
IFI	inspector follow-up item
IP	inspection procedure
k_{eff}	effective neutron multiplication factor
MOX	mixed oxide
NCS	nuclear criticality safety
NCSE	nuclear criticality safety evaluation
NRC	U. S. Nuclear Regulatory Commission
RCW	recirculating cooling water
SAR	Safety Analysis Report
SRI	safety-related item
TEHP	triethyl hexaphosphate
TSR	technical safety requirement
USEC	U. S. Enrichment Corporation