February 26, 2007

Lawrence G. McDade, Chair
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Peter S. Lam
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Richard E. Wardwell
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of
USEC, Inc.
(American Centrifuge Plant)
Docket No. 70-7004

Dear Administrative Judges:

In the "NRC Staff Response to Atomic Safety and Licensing Board Order of February 6, 2007," filed on February 20, 2007 ("Staff Response"), the Staff indicated that the individual NRC staff members who have filed a Differing Professional Opinion (DPO) with regard to the Staff's August 4, 2006 Memorandum might supplement the Staff's response to Board Question S2-1. Staff Response at 23, fn. 11. Enclosed please find the individual NRC staff members' supplement to the Staff Response, along with supporting affidavits signed by Melanie Galloway, Frederick Burrows, and Roman Shaffer. Christopher Tripp is out of the office today due to a family medical emergency and, therefore, was unavailable to sign an affidavit. His affidavit will be filed when he returns to work.

Respectfully submitted,

/RA by Margaret J. Bupp/

Margaret J. Bupp
Counsel for the NRC Staff

Enclosure: As stated

cc w/ encl: D. Wolf
D. Silverman
D. Scott
Office of Commission Appellate Adjudication
Office of the Secretary

February 26, 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

AFFIDAVIT OF FREDERICK H. BURROWS

I, Frederick H. Burrows, do hereby state as follows:

1. I am employed as a senior electrical engineer in the Technical Support Branch in the Division of Fuel Cycle Safety and Safeguards in the Office Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission. In this capacity, I perform electrical/instrumentation reviews associated with regulation of fuel cycle facilities. A brief statement of my professional qualifications is attached.

2. Together with three of my colleagues, I have provided a response to the Board's question S2-1, "Sufficiency of Review Information," attached herewith.

3. I declare under penalty of perjury that the statements made in the attached response are true and correct to the best of my knowledge, information and belief.

Original Signed By:

____
Frederick H. Burrows

Executed in Rockville, MD
this 26th day of February, 2007

February 26, 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

AFFIDAVIT OF MELANIE A. GALLOWAY

I, Melanie A. Galloway, do hereby state as follows:

1. I am employed as Branch Chief of the Technical Support Branch in the Division of Fuel Cycle Safety and Safeguards in the Office Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission. In this capacity, I supervise staff who perform criticality safety reviews and electrical/instrumentation reviews associated with regulation of fuel cycle facilities. A brief statement of my professional qualifications is attached.

2. Together with three of my colleagues, I have provided a response to the Board's question S2-1, "Sufficiency of Review Information," attached herewith.

3. I declare under penalty of perjury that the statements made in the attached response are true and correct to the best of my knowledge, information and belief.

Original Signed By:

____
Melanie A. Galloway

Executed in Rockville, MD
this 26th day of February, 2007

February 26, 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

AFFIDAVIT OF ROMAN A. SHAFFER

I, Roman A. Shaffer, do hereby state as follows:

1. I am employed as an Instrumentation and Controls Engineer in the Instrumentation and Electrical Engineering Branch in the Division of Fuel, Engineering, and Radiological Research in the Office Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission. In this capacity, I am a project manager responsible for research on safety and security issues regarding the use of digital technology in electrical and instrumentation and control systems in nuclear facilities regulated by the U.S. Nuclear Regulatory Commission. Additionally, I have performed instrumentation and control reviews associated with regulation of fuel cycle facilities. A brief statement of my professional qualifications is attached.

2. Together with three of my colleagues, I have provided a response to the Board's question S2-1, "Sufficiency of Review Information," attached herewith.

3. I declare under penalty of perjury that the statements made in the attached response, with the noted exceptions, are true and correct to the best of my knowledge, information and belief.

Original Signed By:

____
Roman A. Shaffer

Executed in Rockville, MD
this 26th day of February, 2007

February 26, 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

AFFIDAVIT OF CHRISTOPHER S. TRIPP

I, Christopher S. Tripp, do hereby state as follows:

1. I am employed as a Sr. Nuclear Process Engineer (Criticality) in the Nuclear Regulatory Commission's Office of Nuclear Material Safety and Safeguards, Division of Fuel Cycle Safety and Safeguards. A brief statement of my professional qualifications is attached.

2. Together with three of my colleagues, I have provided a response to the Board's question S2-1, "Sufficiency of Review Information," attached herewith.

3. I declare under penalty of perjury that the statements made in the attached response are true and correct to the best of my knowledge, information and belief.

____
Christopher S. Tripp

Executed in Rockville, MD
this 26th day of February, 2007

RESPONSE OF FOUR NRC STAFF MEMBERS
TO
QUESTION PERTAINING TO SUFFICIENCY OF REVIEW INFORMATION

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[1], and Staff memoranda of September 13 and October 19, 2006. This discussion should highlight:

1. The applicable[2] sections of 10 CFR Part 70.

Response: The following are applicable sections of 10 CFR Part 70 pertaining to completeness of design:

10 CFR 70.22(a)(7) 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 and 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.).

---

[1]The Staff's (i.e., that portion of the Staff in agreement with the management policy) response (hereafter referred to as the other Staff's response) to this question stated that the August 4, 2006, policy provided the framework for the NRC's review of USEC's license application. However, as shown by the date of the policy memo vs. the issuance of the USEC SER, the policy memo was not written until the technical review of the USEC application was essentially complete and preparations of the SER was in its final stages. In addition, the LES review had been completed, and the license issued when the policy was developed. During the time prior to the memo being written, some of the Staff's technical reviewers were attempting to complete a review of design details but were unable to because of the incompleteness of the application and supporting information. In addition, one of our concerns is that the August 4 memo was issued without any input from technical Staff involved in the review or, as far as we know, endorsement from senior office management.

[2]For our discussion of the applicable sections of 10 CFR Part 70, we divided our answers into two separate, interrelated issues: completeness of design and completeness of the ISA, where appropriate.

10 CFR 70.23(a)(3) states an application for a license will be approved if the Commission determines that:

> The applicant's proposed equipment and facilities are adequate to protect health and minimize danger to life and property.

Although the above two regulations predate Subpart H of 10 CFR 70, we believe that they are interrelated and complement the ISA process in that the equipment, etc., used to protect health and minimize danger to life and property are, to a large degree, determined by the ISA.

The following are applicable sections of 10 CFR Part 70 pertaining to completeness of the ISA:

10 CFR 70.62(c) states that each applicant shall conduct and maintain an ISA, that is of appropriate detail for the complexity of the process, that identifies:

> .
> .
> .
> (v) The consequences and the likelihood of occurrence of each potential accident sequence identified pursuant to paragraph (c)(1)(iv) of this section, and the methods used to determine the consequences and likelihoods.
>
> (vi) Each item relied on for safety identified pursuant to §70.61(e) of the subpart, the characteristics of its preventive, mitigative, or other safety function, and the assumptions and conditions under which the item is relied upon to support compliance with the performance requirements of §70.61.

10 CFR 70.65(b) describes the required contents of the ISA Summary, including:

> (3) A description of each process...analyzed in the integrated safety analysis in sufficient detail to understand the theory of operation; and, for each process, the hazards that were identified in the integrated safety analysis...and a general description of the types of accident sequences.
>
> (4) Information that demonstrates the licensee's compliance with the performance requirements of §70.61, including a description of the management measures...
>
> (6) 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.

Because 10 CFR 70.65(b)(4), in particular, requires demonstration that the applicant has met the performance requirements, the requirements from 10 CFR 70.61(b), (c), (d), and (e) are provided below:

> (b) The risk of each credible high-consequence event must be limited. Engineered controls, administrative controls, or both shall be applied to the extent needed to reduce the likelihood of occurrence of the event so that, upon implementation of such controls, the event is highly unlikely or its consequences are less severe than those in paragraphs (b)(1)-(4) of this section...
>
> (c) The risk of each credible intermediate-consequence event must be limited. Engineered controls, administrative controls, or both shall be applied to the extent needed so that, upon implementation of such controls, the event is unlikely or its consequences are less than those in paragraphs (c)(1)-(4) of this section...
>
> (d) In addition to complying with paragraphs (b) and (c) of this section, the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical...
>
> (e) Each engineered or administrative control or control system necessary to comply with paragraphs (b), (c), or (d) of this section shall be designated as an item relied on for safety. The safety program...shall ensure that each item relied on for safety will be available and reliable to perform its intended function when needed and in the context of the performance requirements.

Because 10 CFR 70.62(c)(v) and (vi); 10 CFR 70.65(b); and 10 CFR 70.61(b), (c), (d), and (e) contain such phrases as "description of <u>each</u> process," "<u>each</u> potential accident sequence," "<u>each</u> item relied on for safety," "<u>each</u> credible high- and intermediate-consequence event," "<u>all</u> nuclear processes are subcritical," and "<u>each</u> engineered or administrative control" needed to comply with the specific regulatory requirements, this means the ISA must be complete. Based on a review of these regulations, a complete ISA is one that identifies all credible high- and intermediate-consequence events, ensures that they all meet the performance requirements, and identifies all controls needed to meet the performance requirements as items relied on for safety (IROFS). (Further guidance on this is provided in NUREG-1520, which is discussed in the memoranda of September 13 and October 19, 2006, and the differing professional opinion (DPO), as well as in our discussion pertaining to Question A.4 herein).

10 CFR 70.66(a) states that a license will be issued if the Commission determines that the applicant has complied with 10 CFR 70.21, 70.22, 70.23, and 70.60 through 70.65.

Our comments on the other Staff's response to this question are contained in our responses to the following questions where appropriate.

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

Response: The differences between the two positions are discussed at length in the DPO and, in particular, the September 13, 2006, memorandum. The key differences are summarized below:

On completeness of design, we believe that 10 CFR Part 70 requires that an applicant must submit enough details in the description of the equipment and facility used to ensure safety, so the Staff's qualified technical reviewers can determine that the equipment and facility are adequate per the regulations and the guidance contained in NUREG-1520 or an acceptable alternative. The level of design detail needed to support licensing must be sufficient for the Staff to have reasonable assurance that all credible scenarios have been identified, that all necessary controls (IROFS) have been identified, and that the risk of all accident sequences is as required to meet the performance requirements.

The level of information needed will vary from one technical discipline to another, but any details which can reasonably be expected to affect safety should be identified. For example, the color and exact model number of a pump may not be known until the facility is constructed, but the flow capacity of the pump, whether it contains hydrocarbon oil of more than a safe volume, and the specific components it is connected to, may be important. Some of the properties of the pump may be important for criticality safety, whereas other properties may be important for chemical safety. NUREG-1520, pages 3-13 through 3-22 discuss the types of information needed to support the Staff's review of the ISA Summary. In our view, this basic level of information is needed as a minimum for the applicant to perform a comprehensive safety analysis of its facilities and processes and for the Staff to reach an acceptable conclusion. This level of detail was not available to us in the USEC application in all cases.

The other Staff's response on completeness of the design is that "the licensing decision is ultimately based on a sufficient level of detail..." But in reality the Staff based (per the August 4 policy) its licensing action on commitments by USEC to follow industry standards and the Staff's inspections required by 10 CFR 70.32(k) to be conducted after issuance of the license.

The other Staff's response to this question stated that the licensing review is programmatic in nature, and uses terms such as "functional-level of design information" and "programmatic and functional commitments."

These terms are being introduced, without being defined, for the first time in the other Staff's response.

Our position is that the licensing review includes both programmatic (reviewing the safety program and ISA commitments required by 10 CFR 70.62) and technical (reviewing the technical information required by 10 CFR 70.22 and 10 CFR 70.23 and the ISA required by 10 CFR 70.62(c) and 10 CFR 70.65) components. Both are required for licensing under 10 CFR 70.66(a). Therefore, the fact that a programmatic review is conducted does not negate the fact that the Staff also has to review detailed technical information to make the findings with regard to Subpart H of 10 CFR Part 70, 10 CFR 70.22(a)(7), and 10 CFR 70.23(a)(3). Our position is supported by the following statement from Section 3.1 of NUREG-1520:

> Reviewers must confirm that an ISA Summary meets the regulatory requirements of 10 CFR 70.65 and, specifically, that suitable IROFS and management measures have been designated for higher-risk accident sequences and that the programmatic commitments to maintain the ISA and ISA Summary are acceptable.

On completeness of the ISA, we believe 10 CFR Part 70 requires that all credible events leading to a high- or intermediate-consequence accident and all IROFS needed to demonstrate that the performance requirements are met must be identified. Also, each and every IROFS must be described in sufficient detail for the qualified technical reviewers to conclude that there is reasonable assurance that the IROFS will make all credible intermediate consequence accidents unlikely, and all credible high consequence accidents highly unlikely.

There were several instances mentioned in the DPO in which the design was not complete and therefore there could be no reasonable assurance that all accident sequences and IROFS were identified. The effect of these incomplete aspects of the design and ISA were that the applicant could not demonstrate for these instances that it had met the performance requirements and could not describe its processes and its IROFS in sufficient detail to satisfy 10 CFR 70.62 and 10 CFR 70.65. Thus, we were unable to determine that the applicant had identified all credible scenarios that can exceed the consequence thresholds in 10 CFR 70.61(b) and (c), and had identified appropriate controls as IROFS to limit risk to an acceptable level as defined by meeting the performance requirements when several accident sequences were identified as not being analyzed and IROFS were not assigned.

The other Staff's response to this question is ambiguous and seems to contradict itself on whether the ISA Summary needs to be complete (contradictions are shown as underlined in the quotation shown below).

> The reasonable assurance standard is applied such that the staff decision pertains to <u>a reasonable assurance that the integrated safety analysis summary is complete</u>....The level of detail required for a licensing decision, therefore, <u>does not require a final facility design or an absolutely complete identification of all items relied on for safety and accident sequences</u>, but instead sufficient information has to be provided to understand the process and functions of items relied on for safety and <u>reasonable assurance that the integrated safety analysis summary is complete</u>.

Based on our interpretation of the above quote, it appears the other Staff is trying to say that the ISA Summary (and ISA) does not need to be complete and that all accident sequences and all IROFS do not need to be identified, which would contradict regulatory requirements as noted above.

The other Staff's response to this question states that the fact that LES was based on an existing facility had no effect on its licensing. Our view is that the Staff obtained considerable benefit from the fact that the LES was based on an existing facility that had operated safely for many years. This provided greater confidence in the design and that scenarios and IROFS associated with these types of operations were understood.

The other Staff's response to this question emphasizes the inspection required by 10 CFR 70.32(k). Our view is that such an inspection is intended to verify that the applicant has constructed the facility in accordance with the license, not to finish the review of the design completed after the license was issued. The design must be reviewed during the licensing stage, before a license can be issued. In addition, currently fuel cycle inspectors do not inspect facilities with the intent of making initial determinations regarding the adequacy of accident scenarios reflecting facility operations and upsets and the adequacy of controls (IROFS) to prevent or mitigate scenario consequences. Instead, inspectors verify that controls, as credited in the ISA, are in place, are able to operate as credited, and are maintained appropriately. The licensing review was also scheduled for eighteen months, whereas a substantially shorter time frame will be allotted for the 10 CFR 70.32(k) inspection.

The other Staff's response to this question also emphasizes the 10 CFR 70.72 change process. Our view is that 10 CFR 70.72 can provide assurance that future changes will be acceptable only if there is a baseline design (including identification of each and every accident sequence and IROFS) against which to measure changes (e.g., does it create a new type of accident sequence, does it remove without an equivalent replacement or alter an IROFS). 10 CFR 70.72 is designed to allow licensees flexibility to make changes to an established design (that has been reviewed and found to be adequate) in a controlled fashion, not to allow applicants to finish designing their facilities after a license has been issued.

No one expects that there will never be any changes to the design. However, at some point in time prior to licensing, the Staff must determine that the applicant has identified all credible scenarios that can exceed the consequence thresholds in 10 CFR 70.61(b) and (c) and has identified appropriate controls as IROFS to limit risk to an acceptable level as defined by meeting the performance requirements (i.e., the Staff must have confidence that the applicant has completed a satisfactory safety assessment of its processes and has met the performance requirements by crediting appropriate IROFS.)

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

Response: The policy in the August 4, 2006, memorandum is inconsistent with the current language in 10 CFR Part 70, as explained in detail in the subject memoranda and the DPO. To make the rule consistent with the policy would require a detailed regulatory analysis that has not yet been done. However, it would appear that implementation of the August 4 memo would involve, for instance, a change to 10 CFR 70.66(a), to allow findings about the completeness of the design and the ISA to be made after the license is issued.

During the technical review, we recognized that aspects of the design had not been completed. As a compromise, we proposed license conditions to require the submission of detailed design information prior to the inspection required under 10 CFR 70.32(k) and suggested that qualified license reviewers be made part of the inspection team. The other Staff determined that the conditions were invalid, because they would have allowed the issuance of a license based on future actions by NRC and that the technical licensing review must be complete prior to licensing. If correct, this would support a need for a rule change. Also, the exact composition of the 10 CFR 70.32(k) inspection team has not been determined.

The other Staff's response to this question included a footnote attempting to draw an analogy between the new inspections, tests, analyses, and acceptance criteria (ITAAC) process for new reactors. No analogue to the ITAAC process has been developed for fuel cycle facilities. Given the diversity of fuel cycle facilities (as compared to the relative similarity between reactors) and the lack of prescriptive design requirements, the need for such a framework may be even greater in fuel facility licensing than for new reactors. This fact argues in favor of a structured process for reviewing new fuel facilities, possibly including a broader rule change than noted above.

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

Response: Our position is consistent with the guidance of NUREG-1520, as discussed at length in the September 13, 2006, memorandum and the DPO.[3] The other Staff's response and the August 4, 2006, policy memo, are inconsistent with the guidance of NUREG-1520 (which has been approved at the Office of the Executive Director (EDO) level and provided to the Commission which is not the case for the policy memo). The most relevant portions of NUREG-1520 are cited below.

With regard to 10 CFR 70.65(b)(3), regarding description of processes, hazards, and types of accident sequences, NUREG-1520 states (emphasis added by underline):

> The description of the processes analyzed as part of the ISA...is considered acceptable if it describes the following features in sufficient detail to permit an understanding of the theory of operation, and to assess compliance with the performance requirements of 10 CFR 70.61. A description at a systems level is acceptable, provided that it permits the NRC reviewer to

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[3]Several years ago staff were assigned to draft a new standard review plan to encompass the Staff's future reviews of gas centrifuge applications. Ultimately, the Division of Fuel Cycle Safety and Safeguards (FCSS) management determined that NUREG-1520 was adequate for application to gas centrifuge licensing, and the separate gas centrifuge SRP effort was curtailed. Late in the USEC gas centrifuge review, FCSS management took a position that NUREG-1520 does not adequately address new facilities such as the gas centrifuges, so the acceptance criteria therein cannot be completely applied. [Note that Roman Shaffer does not have historical knowledge of the preceding statements.] That NUREG-1520 was clearly intended to apply to new facilities is made clear by its title, "Standard Review Plan for the <u>Review of a License Application</u> for a Fuel Cycle Facility," [emphasis added] and the first sentence of the Abstract, which reads: "This 'Standard Review Plan (SRP) for the Review of a License Application for a Fuel Cycle Facility' (NUREG-1520) provides guidance to the Staff reviewers in the U.S. Nuclear Regulatory Commission (NRC)...who perform safety and environmental impact reviews of <u>applications to construct or modify and operate nuclear fuel cycle facilities</u>" [emphasis added]. FCSS management has also stated that the SRP is "just guidance." While it is true that applicants do not have to follow the approaches outlined in the SRP (i.e., they are not regulations), there are repeated statements in NUREG-1520 that if an applicant deviates from the approaches in the SRP, it should provide a justification for how its alternative approach satisfies the regulations. Besides providing one acceptable means of meeting the regulations, the SRP also provides guidance to the Staff as to how to perform its reviews, to ensure regulatory consistency, predictability, and openness. As such, it provides very useful guidance on various regulatory issues and its development involved all stakeholders.

The Safety Evaluation Report (SER) was written to identify where the applicant met the acceptance criteria in NUREG-1520 and where it deviated from NUREG-1520. The SER does not state that the application and technical review substantially deviates from NUREG-1520 but mentions that it is generally applicable with a few exceptions. If NUREG-1520 is not applicable to a new fuel facility, it should not have been used extensively during the review, as documented in the SER. Also, if NUREG-1520 is not applicable, alternative guidance to the Staff should have been provided at the beginning of the review. However, this was not done.

adequately evaluate (1) the completeness of the hazard and accident identification tasks and (2) the likelihood and consequences of the accidents identified....The information provides an adequate explanation of how the IROFS reliably prevent the process from exceeding safety limits for each high and intermediate consequence accident sequence. (page 3-13)

Major components include the general arrangement, function, and operation of major components in the process. If appropriate, it also includes arrangement drawings and process schematics showing the major components and instrumentation, and chemical flow sheets showing compositions of the various process streams." (page 3-13)

Process design and equipment include a discussion of process design, equipment, and instrumentation that is sufficiently detailed to permit an adequate understanding of the results of the ISA. As appropriate, it includes schematics indicating safety interrelationships of parts of the process. In particular, it is usually necessary for criticality safety to diagram the location and geometry of the fissile and other materials in the process, for both normal and bounding abnormal conditions. This can be done using either schematic drawings or textual descriptions indicating the location and geometry of fissile materials, moderators, etc., sufficient to permit an understanding of how the IROFS limit the mass, geometry, moderation, reflection, etc. (page 3-13)

Process operating ranges and limits include the operating ranges and limits for measured process variables (e.g., temperatures, pressures, flows, and compositions) that are controlled by IROFS to ensure safe operations of the process. (page 3-13)

The description of process hazards provided in the ISA Summary is acceptable if it identifies, for each process, all types of hazards that are relevant to determine compliance with the performance criteria of 10 CFR 70.61. That is, the acceptance criterion is completeness. All hazards that could result in an accident sequence in which the consequences could exceed the performance requirements of 10 CFR 70.61 should be listed...Otherwise the reviewer(s) cannot determine completeness. (page 3-13)

The list of process hazards is acceptable if the ISA Summary provides the following information...a list of materials (radioactive, fissile, flammable, and toxic) that could result in hazardous situations (e.g., loss of containment of licensed nuclear material), including the maximum intended inventory amounts and location(s) of the hazardous materials at the facility...potential interactions among materials or conditions that could result in hazardous

> situations. (page 3-14)
>
> The general description of types of accident sequences in the ISA Summary is acceptable if the reviewer can determine the following considerations...The applicant has identified <u>all accidents</u> for which the consequences could exceed the performance requirements of 10 CFR 70.61...The applicant has identified how the IROFS listed in the ISA Summary protect against <u>each such type of accident</u>. (page 3-14)
>
> ...it is not generally acceptable to merely list the type of hazard or the controlled parameters without referencing the items relied on to control that parameter or hazard. The description of general types of accident sequences is acceptable if it covers <u>all types of sequences, initiating events and IROFS failures</u>...The description of a general type of accident sequence is acceptable if it permits the reviewer to determine how <u>each accident sequence</u> for which the consequences could exceed the performance requirements of 10 CFR 70.61 is protected against by IROFS or a system of IROFS. (page 3-14)
>
> <u>To demonstrate completeness</u>, the description of general types of accident sequences must be identified using systematic methods and consistent references. Therefore, each description of a general type of accident sequence is acceptable if it meets the following criteria....<u>The applicant did not overlook any accident sequence for which the consequences could exceed the performance requirements of 10 CFR 70.61</u>. (page 3-14 to 3-15)

The above excerpts make clear that fairly detailed technical information are needed to make regulatory conclusions pertaining to the description of processes, hazards, types of accident sequences, and IROFS. The phrase “as appropriate” means that the exact type of information needed to draw conclusions in each safety discipline may vary, depending on the situation. The individual reviewer is best suited to determine the exact information needed for doing this. The excerpts also make it clear that the expectation is completeness of the accident sequences and the associated IROFS and that this determination can only be made with a fairly detailed level of design information.

The other Staff’s response to this question is not consistent with this guidance, in that it makes an *a priori* judgement that the design was sufficient, based on generic programmatic commitments (such as a generic commitment to industry standards), even though the technical reviewers concluded there was not sufficient design information in some instances. It also is inconsistent with NUREG-1520 because it concludes that a licensing decision “does not require a final facility design or an absolutely complete identification of all items relied on for safety and accident sequences” (August 4, 2006, memo).

With regard to 10 CFR 70.65(b)(4), regarding demonstrating compliance with the performance requirements, NUREG-1520 states:

> The performance requirements of 10 CFR 70.61 have three elements, including (a) <u>completeness</u>, (b) consequences, and (c) likelihood. Completeness refers to the fact that the ISA must address *each* [NOTE: This word is italicized in the original.] credible event. (page 3-15 to 3-16)
>
> <u>Completeness</u> is demonstrated by correctly applying an appropriate accident identification method...Specific acceptance criteria for completeness are covered in item 3 above. (page 3-16)

The above excerpts make clear that demonstration that an applicant has met the performance requirement entails identifying all credible accident sequences that can cause high- or intermediate-consequence events. Also, a programmatic commitment to follow an accident identification methodology is not enough; the Staff must find that the applicant has correctly applied that method.

With regard to 10 CFR 70.65(b)(6), regarding descriptions of IROFS, NUREG-1520 states:

> The "list describing items relied on for safety"...is acceptable, provided the following conditions are met...<u>The list includes *all*</u> [NOTE: This word is italicized in the original.] <u>IROFS</u> in the identified high and intermediate consequence accident sequences. (page 3-21)
>
> The *description* of the IROFS includes management measures applied to the IROFS (including the safety grading), characteristics of its preventive, mitigative, or other safety function, and assumptions and conditions under which the item is relied on to support compliance with the performance requirements of 10 CFR 70.61. If information on any safety limits and safety margins associated with an IROFS is not provided in the ISA Summary, <u>it must be available for review in ISA documentation onsite</u>. (page 3-21)
>
> The above acceptance criteria are explained in greater detail below:
>
>> All items: The primary function of the list describing each IROFS is to document the safety basis of all processes in the facility...<u>the key feature of this list is that *all*</u> [NOTE: this word is italicized in the original.] <u>IROFS are included. To be acceptable, no item, aspect, feature, or property of a process that is needed to show compliance with the safety performance requirements of the regulation may be left off</u>

> this list...The ISA Summary need not provide a breakdown of hardware IROFS by component or identify all support systems. However, the ISA documentation maintained on site, such as system schematics and/or descriptive lists, should contain sufficient detail about items within a hardware IROFS, such that it is clear to the reviewer(s) and the applicant, what structure, system, equipment, or component is included within the hardware IROFS' boundary...[4] (page 3-21)
>
> Description of Items: The essential features of each IROFS should be described. Sufficient information should be provided about engineered hardware controls to permit an evaluation that, in principle, controls of this type will have adequate reliability. Because the likelihood of failure of items often depends on safety margins, the safety parameter controlled by the item, the safety limit on the parameter, and the margin to true failure should, in general, be described. For IROFS that are administrative controls, the nature of the action or prohibition involved must be described sufficiently to permit an understanding that, in principle, adherence to it should be reliable. Features of the IROFS that affect its independence from other IROFS, such as reliance on the same power supplies, should be indicated. (page 3-21 to 3-22)
>
> The description of each IROFS should identify its expected function, conditions needed for the IROFS to reliably perform its function, and the effects of its failure. The description of each IROFS within an ISA Summary should identify what management measures...are applied to it. (page 3-22)

The above excerpts pertaining to the descriptions of IROFS make clear that all IROFS must be described and must be described in sufficient detail to permit the Staff to conclude the applicant has met the performance requirements of 10 CFR 70.61. Pages 3-21 to 3-22 of NUREG-1520 contain lengthy guidance on the level of detail needed for both engineered and administrative IROFS, besides what is reproduced above.

Again, the other Staff's response to this question is not consistent with this guidance in NUREG-1520 (nor with regulatory requirements), in that it states that a licensing decision "does not require a final facility design or

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[4]This paragraph and the one that follows it are essentially the appropriate standard for the acceptable level of design information and the completeness of the design and ISA as pertaining to IROFS.

an absolutely complete identification of all items relied on for safety and accident sequences" (August 4, 2006, memo).

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

Response: We do not believe that relying on programmatic commitments in lieu of appropriate design detail is supported by the rule (10 CFR Part 70) or existing regulatory guidance (NUREG-1520) (as justified in our previous responses).

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.

Response: We believe that the current rule (10 CFR Part 70) and regulatory guidance (NUREG-1520) require the Staff to review appropriate design detail and a complete ISA based on that design before a license may be issued. The inspection required under 10 CFR 70.32(k) will provide an opportunity to verify that the facility as built will comply with the license application and will be consistent with the ISA Summary. Changes to the design and operation of the facility following issuance of the license will be subject to the 10 CFR 70.72 change process. These two requirements are part of the existing overall licensing framework and function together with the licensing review to provide reasonable assurance of safety during the life of the facility.

However, these regulatory requirements (10 CFR 70.32(k) and 10 CFR 70.72) that take effect following the issuance of the license will only provide the necessary confidence if the Staff is able to complete a technical review (as described in NUREG-1520) of a complete ISA based on appropriate design detail. Additional inspection resources, time, and inspector training would be required to permit the inspectors to expand their current role to include the necessary detailed design review (as the license reviewers do now). Questions about how 10 CFR 70.72 would be applied without a sufficiently complete baseline design (e.g., all accident sequences and all IROFS identified) against which to measure changes would have to be answered. We believe 10 CFR 70.72 is intended to allow licensees latitude to make changes to their facilities and processes which had been thoroughly reviewed and found acceptable by the Staff during the licensing review. The NRC's review of 10 CFR 70.72-controlled design changes is not a substitute for a complete design review during licensing.

Options to address the current situation could include: (1) inclusion of a license condition similar to that originally proposed, to require submission of detailed design information prior to the required inspection, to allow the technical Staff time to review it; (2) referral of the application back to the Staff for additional technical review, as more detailed design information becomes available; (3) changes to 10 CFR Part 70 (especially 70.66(a)) to change the standard required for licensing; or (4) the issuance of an exemption under 10 CFR 70.17 (if warranted) to provide a basis to allow licensing even though all of the requirements listed in 10 CFR 70.66(a) have not been met. In the DPO, we asked the panel to provide possible alternatives for resolving the inconsistency between the August 4, 2006, policy memo and 10 CFR Part 70 and NUREG-1520.

Statement of Professional Qualifications
of Frederick H. Burrows

Experience:

U.S. Nuclear Regulatory Commission (USNRC)

Electrical Engineer, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards
February 2000 to present

- Review and evaluate electrical power and instrumentation and control (I&C) systems needed for the safe operation of fuel cycle facilities per the NRC regulatory program. Perform engineering reviews which include conditions arising from normal operation and postulated events/accidents, including surveillance and maintenance.

Electrical Engineer, Division of Engineering, Office of Nuclear Reactor Regulation
July 1988 to February 2000

- Reviewed and evaluated electrical power systems and associated instrumentation and controls needed for the safe operation and shutdown of nuclear power plants per the NRC regulatory program. Performed engineering reviews which included conditions arising from normal operation and postulated accidents, including surveillance and maintenance issues and operating reactor problems/events.

Electrical Engineer, Division of Engineering, Office of Nuclear Reactor Regulation
May 1981 to July 1988

- Reviewed and evaluated (per NRC the regulatory program) instrumentation and control systems which provide automatic protection and control against unsafe reactor operation during steady-state and transient power operations and which provide initiating signals and proper control of safety systems to mitigate the consequences of accident conditions.

Other Experience:

- Electrical/Electronics/General Engineer, Naval Sea Systems Command, Department of the Navy
- Electrical Engineer, Northern Indiana Public Service Company

EDUCATION:

- B.S.E.E. - Valparaiso University - 1968
- M.S.E.E. - University of Notre Dame - 1971

MILITARY SERVICE:

- Electronics Technician, U.S. Navy, 1960-1964

AWARDS:

- Senior Honors---Valparaiso University, 1968
- High Quality Performance---1987, 1993, 1995 (NRC)
- Special Performance---1997, 1998, 2005 (NRC)
- Special Act---1998, 1999, 2005 (NRC)
- Instant Cash---1996, 2000, 2004 (NRC)

Statement of Professional Qualifications
of Melanie A. Galloway

Experience:

U.S. Nuclear Regulatory Commission (USNRC)

Chief, Technical Support Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards
February 2004 to present
- Responsibilities include fuel cycle inspection program, criticality safety inspection function, criticality and electrical/instrumentation and control safety reviews to support licensing actions, integrated safety assessment policy, and risk-informed regulatory applications to fuel cycle

Acting Deputy Director, Division of Inspection and Regional Support, Office of Nuclear Reactor Regulation
June 2006 - August 2006
- Responsible for two branches that oversee the inspection program and the reactor oversight process including the significance determination process and the performance indicator program

***Technical Assistant for Security*,** FCSS, NMSS, USNRC
July 2002 - February 2004
- Technical manager for nearly $3 million contract with Sandia National Laboratory for performance of security assessments for fuel cycle, materials, and waste facilities. NRC lead for the joint Department of Energy/NRC report on radiological dispersal devices which defined radionuclides of greatest concern and thresholds.

Section Chief, Rulemaking Section A, Division of Industrial and Medical Nuclear Safety, NMSS,
May 2001 - June 2002
- Responsible for the development of complex rulemakings, maintenance of regulatory guidance and inspection procedures, and analysis of Office of Management and Budget information collection requirements.

***Section Chief, Enrichment Section*,** FCSS, NMSS, USNRC
September 1998 - April 2001
- Responsible for adequacy of safety and safeguards controls and adherence to NRC requirements at the Portsmouth and Paducah Gaseous Diffusion Plants. Developed the licensing framework for the Mixed Oxide Fuel Fabrication Facility. Completed first recertification of the GDPs and a report to Congress on GDP operation.

***Section Chief, Licensing Section 2*,** FCSS, NMSS, USNRC
September 1997 - September 1998
- Responsible for performing safety and safeguards reviews for commercial fuel fabrication facilities and source material processes. Developed first draft of Part 70 Standard Review Plan.

Acting Branch Chief, Division of Reactor Projects, Region III
July - August 1997
- Managed regional oversight of nuclear power plants, including two on NRC's watch list.

Senior Emergency Response Coordinator, Incident Response Division, Office for Analysis and Evaluation of Operational Data, USNRC
August 1995 - July 1997
- Developed concept, plan, and procedures for News Center of Operations Center. Member of Federal Emergency Management Agency's Strategic Review Steering Committee for review of Radiological Emergency Response Program.

Other Experience:
- Acting Senior Project Manager, Office of Nuclear Reactor Regulation
- Senior Reactor Operations Engineer, Office for Analysis and Evaluation of Operational Data
- Acting Branch Chief, Region IV
- Reactor Operations Engineer, Office of Nuclear Reactor Regulation
- Acting Resident Inspector, Calvert Cliffs, Region I
- Project Manager, TENERA, LLP
- Technical Assistant, Office of Nuclear Reactor Regulation
- Project Manager, Office of Nuclear Reactor Regulation
- Summer Technical Intern, Savannah River Plant, DuPont

Education:
- B.S., Nuclear Engineering, The Pennsylvania State University, 1982
- M.B.A., University of Maryland, 1988

Awards, Honors, and Meritorious Selections:
- Senior Executive Service Candidate Development Program, February 2006 to present
- Meritorious Service Award, June 2000
- High Quality Increases, September 1987 and December 1996
- Recent Performance Awards, December 1999, December 2001, February 2003, August 2003, December 2004, December 2005, December 2006
- American Political Science Association Congressional Fellow, October 1992 - August 1993
- Executive Potential Program for Mid-Level Employees, March 1996 - March 1997
- Eric A. Walker Award, The Pennsylvania State University, one of two highest awards to graduating senior, 1982

Training Highlights:
- Strategic Management of Regulatory and Enforcement Agencies, Harvard Kennedy School of Government, October 2006
- Leadership for a Democratic Society, Federal Executive Institute, March 2000
- Human Resources Management Practices, NRC, September 1999
- Environmental Policy Issues Seminar, Eastern Management Development Center, March 1999
- Supervising Human Resources, NRC, December 1997

Statement of Professional Qualifications
Roman A. Shaffer

WORK EXPERIENCE
June 2000 - Present
Instrumentation & Control Engineer, GG-14
United States Nuclear Regulatory Commission/Office of Nuclear Regulatory Research

- Research Project Manager in the areas of cyber security, fault-tolerant design, and software safety and risk
- Technical reviewer for the safety assessment of the instrumentation and controls systems for both the Louisiana Energy Services National Enrichment Facility planned for construction at Eunice, New Mexico, and the USEC, Inc., American Centrifuge Plant planned for construction at Piketon, Ohio

September 1997 - May 2000
Research Assistant, Teaching Assistant
The Pennsylvania State University, State College, PA

February 1988 - February 1994
Nuclear Reactor Operator; Electronics Technician; Reactor Technician
United States Navy, U.S.S. Abraham Lincoln, CVN-72

EDUCATION
May 2000
Master of Science in Nuclear Engineering
The Pennsylvania State University
State College, PA

September 1997
Bachelor of Science in Electrical Engineering
Bachelor of Science in Computer Engineering
Colorado Technical University
Colorado Springs, CO

AWARDS

- Pennsylvania State University: Institute of Nuclear Plant Operators Fellowship, Dean's Fellowship
- Colorado Technical University: Dean's Honor Roll, Dean's Academic Achievement Award, Colorado Merit Scholarship Award

Statement of Professional Qualifications
Dr. Christopher S. Tripp

Work Experience

Nuclear Criticality Safety License Reviewer 1998-2007
USNRC, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards

Responsibilities as a Senior Nuclear Process Engineer (GG-15) include technical review of license applications, amendment requests, and integrated safety analyses; policy and guidance development; event response; and inspection and enforcement activities. This also involved an eighteen-month period as Acting Team Leader for the Criticality Team and a six-month rotation to the Spent Fuel Project Office.

Major work assignments include being the senior nuclear criticality safety reviewer for the following: USEC American Centrifuge Plant, Mixed-Oxide Fuel Fabrication Facility, Paducah Gaseous Diffusion Higher Assay Upgrade Project, and Nuclear Fuel Services KAST Project. Assignments also include numerous license renewals and amendments, including technical support on the license application for Louisiana Energy Services, the Atomic Vapor Laser Isotope Separation review, and the external regulation pilot for Oak Ridge's Radiochemical Engineering and Development Center. Reviewed the Integrated Safety Analyses for Nuclear Fuel Services, the Mixed-Oxide Fuel Fabrication Facility, and USEC American Centrifuge Plant. Prepared Chapter 6 and Appendix A of the Mixed-Oxide Standard Review Plan (NUREG-1718), which is based on the Part 70 Standard Review Plan (NUREG-1520). Authored or co-authored interim staff guides FCSS-ISG-01, FCSS-ISG-03, and FCSS-ISG-10.

Nuclear Criticality Safety Inspector 1996-1998
USNRC, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards

Responsibilities included conducting inspections at enrichment and fuel fabrication facilities (including both low- and high-enriched uranium facilities), as well as associated event response and enforcement activities.

Research and Teaching Assistant 1990-1995
Rensselaer Polytechnic Institute, Department of Physics

Responsibilities included participating in a multi-disciplinary team on two experiments at the DOE-MIT Bates Linear Accelerator Center, including analysis of the experimental results and preparing a doctoral thesis based thereon, as well as teaching undergraduates in the Physics Department.

Education

B.S. in Physics, Rensselaer Polytechnic Institute	1989
M.S. in Physics, Rensselaer Polytechnic Institute	1994
Ph.D. in Physics, Rensselaer Polytechnic Institute	1995
Nuclear Criticality Safety Inspector Qualification, USNRC	1997
Nuclear Criticality Safety License Reviewer Qualification, USNRC	1999

Awards

Sigma Pi Sigma National Physics Honor Society	1988
Performance Award (NFS KAST Amendment)	1999
Annual Performance Awards	2004-2006
Arthur S. Flemming Award	2006