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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

RAS 13443

March 16, 2007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Docket No. 70-7004-ML

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 TESTIMONY RELATED TO HTS-12: ISA AND ISA SUMMARY OF REVIEW INFORMATION (S2-1).

Q1: Please state your name, occupation, by whom you are employed, and your professional qualifications.

A1: (TJ) My name is Timothy Johnson. I am employed as a Senior Project Manager in the NRC's Office of Nuclear Materials Safety and Safeguards, Division of Fuel Cycle Safety and Safeguards. A statement of my professional qualifications has been previously provided.

A1: (JH) My name is Jay Henson. I am a Branch Chief in the NRC's Region II office, Division of Fuel Facility Inspection. A statement of my professional qualifications has been previously provided.

A1: (BS) My name is Brian W. Smith. I am the Chief of the Enrichment and Conversion Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards. A statement of my professional qualifications is attached.

A1: (RW) My name is Rex Wescott. I am employed as a Senior Fire Protection Engineer in the NRC's Office of Nuclear Materials Safety and Safeguards, Division of

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SECY-02

Fuel Cycle Safety and Safeguards. A statement of my professional qualifications has been previously provided.

Q2: Please describe your professional responsibilities with regard to the NRC staff's ("Staff") review of the USEC, Inc.'s ("the Applicant") license application ("Application") for the proposed American Centrifuge Plant (ACP) in Piketon, Ohio.

A2: (TJ) I reviewed the information provided by the Applicant in connection with their decommissioning funding plan and prepared Chapter 10.3.2 of the SER.

A2: (JH) If the Applicant is granted a license, I will supervise the Operational Readiness Review (ORR) inspection that must be completed before the Applicant can begin operations and will also supervise regular facility inspections during operation of the ACP.

A2: (BS) As Branch Chief, I was the first-line manager responsible for supervising the Staff's completion of the Safety Evaluation Report (SER) for the proposed ACP. NUREG-1851, "Safety Evaluation Report for the American Centrifuge Plant in Piketon, Ohio" (2006), Staff Exhibit 1.

A2: (RW) I was the Fire Safety and Integrated Safety Analysis Reviewer for the Staff's review of the USEC Application.

<u>HTS-12: ISA and ISA Summary: Sufficiency of Review Information (S2-1).</u>

Q3: The NRC Staff stated that the ISA Summary shall contain a description of "each process" (defined as a single reasonably simple integrated unit operation within an overall production line (10 CFR § 70.65(b)(3)) analyzed in sufficient detail to understand the theory of its operation. The Staff then concluded that a functional-level of design information is sufficient for this review. The Staff then seemingly went on to state that, in its judgment, the description of the programmatic provisions of USEC's proposed activities are adequate for this functional review. Elaborate on what the Staff means by a programmatic review, providing multiple examples of the nature and depth of this review, as compared to a design-depth review, and specifically illustrating how this level of review meets the functional review criteria.

A3: (TJ) In 10 CFR Part 70 licensing, the Staff uses a reasonable assurance standard and focuses on the programmatic provisions of the applicant's proposed activities. By "programmatic," the Staff means descriptions of safety and administrative programs (e.g., health physics program, quality assurance program, etc.) and, as used in the August 4, 2006, memorandum from Robert Pierson, "United States Enrichment Corporation License Detail Regarding the Level of Information Needed for 10 CFR Part 70 Licensing," (Staff Exhibit 60), it also means a level of review of structures, systems, and component designs at a functional level as opposed to a detailed, final design-level construction specification level (e.g., implementation of component and system designs through code and standard programs). The functional design level of review is reflected in the licensing requirements in 10 CFR §§ 70.65(b)(3) and 70.65(b)(6), which refer to: "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."

In the August 4, 2006, Pierson memorandum (Staff Exhibit 60), "programmatic" refers to an applicant's programmatic commitment to use specific codes and standards in the design of structures, systems, and components of the facility. This is also reflected in the various chapters of the standard review plan, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." Based on this understanding, the licensing review needs to focus on the applicant's functional-level commitments. Consequently, the licensing decision is ultimately based on a sufficient level of detail to understand process system functions and how items relied on for safety can perform their intended function and be reliable.

In SECY-00-0111, the NRC Staff discussed the level of detail needed in the integrated safety analysis summary in the final rulemaking that promulgated the 10 CFR Part 70, Subpart H requirements. Memorandum to the Commission from William D. Travers, "Evaluation of the Feasibility of the NRC Creating and Maintaining a Web Page Serving as a Bulletin Board for Agreement State Rulemaking Activities, May 18, 2000 (Staff Exhibit 55 at Attachment 6, page 4). One of the comments to the proposed rule was that items relied on for safety should be described at a "systems level," rather than "in sufficient detail to understand their functions in relation to the performance requirements." In response to this comment, no changes to the rule language were made, because the Staff believed that the current language allowed for the description of items relied on for safety at a "systems level," and because the Staff believed it is necessary to understand the functions of the items relied on for safety.

The level of technical detail necessary for the Staff to assess the effectiveness and reliability of controls (including items relied on for safety) depends on the degree to which the use of the 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 effectiveness and reliability are dependent on management measures such as inspection, testings, and maintenance. The use of industry experience and the existence of relevant codes / standards to facilitate Staff review.

Many of the controls proposed to be used in the ACP have been similarly used before in the gaseous diffusion plants and the Lead Cascade. Hence, their effectiveness and reliability are known to the Staff 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 standard passive controls or administrative controls (e.g., pressure vessels or combustible loading controls) do not require detailed descriptions. For more complex, but standard systems, such as fire or gas detection systems a detection and alarm system or fire suppression systems, the design basis bases of the system will allow the Staff to evaluate their effectiveness. The reliability of most types of standard systems can be assured through appropriate codes and standards. Both types of systems will be assured though industry standards.

The regulations in 10 CFR § 70.65(b)(3) require that the integrated safety analysis summary contain:

> "A description of each process (defined as a single reasonably simple integrated unit operation within an overall production line) 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 pursuant to § 70.62(c)(i) - (iii) and a general description of the types of accident sequences."

The Staff interprets the phrase, "in sufficient detail to understand the theory of operation," to mean sufficient detail to understand the process system functions and functionally how the items relied on for safety can perform their intended function and be reliable. By programmatic, the Staff means that programmatic commitments to, for example, the use of codes and standards for detailed design, fabrication, inspection, and testing provisions are acceptable.

For example, in Section 3.6.4 of the Applicant's ISA Summary, the Applicant describes its process for sampling and transfer. ISA Summary at 3-22 – 3-24. This process includes the autoclaves, evacuation cold traps, accountability scales, vents, sample manifolds, and the transfer manifold. The description in the integrated safety analysis summary includes discussion of the functions of individual components for this process system. One of the key components in this system is the piping system for liquid UF_6 and is listed as an item IROFS on Table 7.2-1 under Liquid Primary System Integrity in the ISA Summary. Staff Exhibit 61. This system piping provides containment of liquid UF_6 during sampling and transfer operations. In Section 7.3.4.13 of ISA Summary, the Applicant discusses the functions of this item relied on for safety and its attributes, which include design and installation in accordance with American Society of Mechanical Engineers (ASME) B31.3, "Process Piping." ISA Summary at 7-40 – 7-41. For evaluating the pressure integrity of this piping system, it is sufficient to state that the design and installation provisions of ASME B31.3 will be used. A level of design detail that states the sizes of the pipe, the piping thicknesses, how piping sections are welded or joined together, what procedures will be used for non-destructive examination, and so forth, would be unnecessary as the ASME B31.3 code prescribes how to determine these "design-depth" details based on the functional design goals of the system (for example, system pressures and temperatures). In addition, the Applicant's implementation of the ASME B31.3 code provisions can be readily inspected by NRC inspection staff and would not require further "licensing review." Therefore, the function of the above piping system is to provide containment of liquid uranium hexafluoride during operations. The containment function is achieved through the proper design and fabrication requirements set out in the applicant's programmatic commitment to use ASME B31.3.

Another example is the process systems using uranium hexafluoride cylinders. These systems are the feed, withdrawal, and sample and transfer systems described in Sections 3.6.1, 3.6.3, and 3.6.4 of the ISA Summary. ISA Summary at 3-7 – 3-10, 3-19 – 3-21, 3-22 – 3-24. In Section 7.3.4.12 of the ISA Summary, the applicant committed to use UF_6 cylinders in accordance with American National Standards Institute (ANSI) N14.1, "American National Standard for Nuclear Materials - Uranium Hexafluoride - Packaging for Transportation," to ensure integrity of the cylinders and prevent releases of UF_6 during operations and introduction of a moderator into the cylinder to reduce the potential for criticality. ISA Summary at 7-40. This section describes the function of the cylinders and their attributes. The cylinders are listed as an IROFS in Table 7.2-1 under Cylinder Integrity Specifications. Staff Exhibit 61. The ANSI N14.1 standard prescribes requirements for cylinder design, fabrication, in-service inspection, maintenance, and testing. For licensing, it would be unnecessary to describe the details of the UF_6 cylinder design and use, because these details are prescribed in the standard. The application of the ANSI N14.1 standard can be readily inspected by NRC inspection staff to ensure that UF_6 cylinders are properly designed and used by the licensee. Therefore, the function of the UF_6 cylinders is to provide containment of UF_6 during operations and to prevent the introduction of a moderator. The containment function is achieved through the proper design and use requirements set out in the programmatic commitment to use ANSI N14.1.

In the area of fire safety, the Applicant has made programmatic commitments to National Fire Protection Association (NFPA) codes for most aspects of its fire protection program and design starting with NFPA 801 "Standards for Fire Protection for Facilities Handling Radioactive Materials." *See* LA at 1.4.6. This code provides prescriptive guidance on Administrative Controls, General Facility Design, General Fire Protection Systems and Equipment, and Special Hazards. The Applicant also has committed to specific codes such as NFPA 13 "Standard for the Installation of Sprinkler Systems." In reviewing the automatic fire suppression system in accordance with NFPA 13, the Staff reviews the design bases of the system (density of water coverage) in accordance with the use of the area and the projected combustible loading. The staff may also review the firewater distribution system in relation to pressure, flow, and reliability to assure an adequate supply to water based fire suppression systems. Inspections during and after construction will assure that NFPA 13 code specifications such as sprinkler spacing, pipe installation, sprinkler locations, and alarm interfaces are followed.

Programmatic administrative controls are also used as items relied on for safety in the area of fire safety. A programmatic item relied on for safety credited by the applicant as preventing a UF_6 cylinder rupture due to a fire in the cylinder yard or during on-site transportation is Fire Department Response. This item relied on for safety is expected to be available along with other preventive items relied on for safety, such as Cylinder Yard Fire Suppression, Cylinder Handling Equipment Fire Suppression System, Cylinder Handling Equipment Use Restrictions, Cylinder Yard grade, and Concrete Cylinder Storage Areas. In order to perform its intended function, the Fire Department Response must be capable of applying water to the fire (through a high pressure hose stream) within a certain minimum time interval with a specified reliability (20 minutes at 90% reliability). The time interval was established through USEC studies of fire induced cylinder ruptures and agreed to by the NRC staff from other independent studies including one cited in the LES review. The reliability was based on data from USEC fire department training exercises. The effectiveness and reliability of the fire department will be assured by adherence to Section 11.3.1.12.1 of the License Application (Fire Protection and Emergency Management Training) which commits to maintaining state certification requirements for firefighters.

In conclusion, Staff considers, as is demonstrated by the examples cited above, that the applicant provided a sufficient level of detail in its application and integrated safety analysis summary for the Staff to perform a licensing review to the reasonable assurance standard and reach the conclusions presented in the Staff's Safety Evaluation Report. Staff Exhibit 1.

Q4: The Staff noted that in order to assure that an applicant's programs have been sufficiently implemented and that commitments have been properly applied in the final facility design and in the constructed facility, "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."

Will the Staff have additional opportunities to review the final design prior to construction?

A4: (JH, BS) No. Construction is expected to commence soon after USEC receives a license. Additional design detail is expected to be completed over the expected 2 years of construction prior to the initial plant operation.

Q5: Will the Staff have additional opportunities to review the final design during construction of the facility?

A5: (JH, BS) Yes, the staff will have several opportunities. If USEC receives a license in April of this year, construction is expected to commence soon afterwards. USEC recently announced that it expects to start commercial plant operations in late 2009. News Release, "USEC Updates Cost Estimate and Schedule for American Centrifuge Plant," February 12, 2007 (Staff Exhibit 11).

Licensees may make changes to their facilities without NRC prior approval. *See* 10 CFR § 70.72(c). The regulations in 10 CFR § 70.72 require licensees to submit to the NRC annually, within 30 days after the end of the calendar year during which these changes occurred, a brief summary of all changes to the records required by 10 CFR § 70.62(a)(2) and for all changes that affect the ISA Summary, licensees are required to submit revised ISA Summary pages. See 10 CFR §§ 70.72(d)(2), (3).

Were the license to be issued in 2007, USEC would be required to submit its facility changes and any ISA Summary page changes to NRC in 2008 and 2009. Consistent with its process for other existing licensees, the staff plans to review these submissions. However, because a potentially significant number of changes is expected, the staff has budgeted extra FTE for these reviews and also plans to conduct an on-site review of the ISA information. In addition to these submissions, USEC has committed to provide to the NRC 180 days prior to the introduction of UF_6 in the American Centrifuge Plant a revised ISA Summary that incorporates all changes that have occurred since the issuance of the materials license. As USEC completes its design, they may make changes to its facility that, under 10 CFR § 70.72, require amendments to be submitted. These amendments would be reviewed by the Staff. Although USEC can proceed with construction of these design changes prior to the Staff's completion of the amendment review, it would have to comply with any requirements issued by the staff in the approved license amendment.

Q6: Will the Staff be performing any inspections during construction of the facility?

A6: (JH, BS) Yes. In accordance with 10 CFR §§ 40.41(g) and 70.32(k), the NRC must ensure that no person 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. As a part of its effort to verify that a uranium enrichment facility has been constructed as required, the NRC must verify in the inspection that the final designs of, including any design changes to, safety significant structural features, equipment, and components comply with the license and the regulations. The NRC must also verify that these structural features, equipment, and components have been constructed and installed as designed. The NRC will also verify in these inspections that both hardware and administrative IROFS have been incorporated into the facility in accordance with the license commitments.

NRC Region II staff, with assistance from NMSS and contractor staff, will inspect, on a prioritized sample basis, the construction of the facility designs using a modified version of IMC 2696, "Louisiana Energy Services (LES) Gas Centrifuge Facility Construction and Pre-Operational Readiness Review Inspection Programs." It is being modified to recognize some of the differences in LES and the ACP, such as the requirement to inspect certain refurbished equipment and structural components at the ACP as well as certain newly constructed structures, equipment, and components. These inspections are performed to ensure the licensee has adequately implemented programs, processes and procedures (e.g., Quality Assurance Program) to design, construct, install and test the structures, equipment, and components that are necessary to protect the health and safety of workers, the public and the environment as required by NRC regulations and the license. NRC inspectors will also select certain IROFS and other safety significant structures, equipment and components for an in-depth assessment of the design, construction, installation and testing of these items.

The NRC's construction and operations inspection program includes a high level review and assessment of the licensee's conduct and maintenance of the ISA and an in-depth review and assessment of selected elements of the ISA (e.g., certain 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. This will include inspections of selected IROFS boundary packages as a part of the design review process and inspections of the installation of selected IROFS related equipment, components, and procedures prior to plant operation. These in-depth inspections will be focused on criticality safety, radiation safety, fire safety, and chemical safety aspects of operations.

USEC will be required to follow its procedure for defining the boundaries of each of its IROFS. These boundaries are required to be available for inspection at the time of the operational readiness review. Staff Exhibit 1 at A-37. Although the operational readiness review will be conducted close to the time of plant operation, the Region staff will be conducting inspections during construction of the facility. As discussed above, the Region staff will inspect IROFS boundary packages during these inspections. Based on the proposed construction schedule, the Region will coordinate with USEC on the availability of certain IROFS boundary packages such that they can be inspected prior to construction of those IROFS. The purpose of these reviews is to ensure that the IROFS are consistent with commitments made in the LA (e.g., compliance with various codes and standards, consistent with design bases) and the assumptions made in the ISA Summary. Once it is determined that the design is consistent with the LA, the Region will inspect the construction of the facility to ensure that it is consistent with the IROFS boundary package.

During the pre-operational inspection phase (operational readiness review), NRC inspectors will also perform risk-informed and performance-based inspections across key functional areas that include areas such as chemical safety, fire protection, radiological control programs, emergency preparedness, training and qualification of plant personnel, and criticality safety. These inspections ensure that the licensee has established and implemented the policies, programs, and procedures important to the safe operation of the facility. Existing procedures described in IMC 2600, "Fuel Cycle Facility Operational Safety and Safeguards Inspection Program," will be used to perform inspections in each program area. This is the IMC that applies to the inspection of licensed activities when the facility begins operation. In addition, as with operating facilities, the licensee's performance during the construction and pre-operational phase will be reviewed as described in IMC 2604, "Licensee Performance Review." The results of the construction and pre-operational inspections will be used to support the NRC's decision regarding USEC's readiness to safely operate the ACP.

Q7: Since these reviews will take place during construction, what happens if design differences are identified?

A7: (JH, BS) It is the responsibility of the licensee to complete its design and construct its facility in accordance with the commitments made in its license application and the assumptions made in its ISA Summary (i.e., original design). As USEC completes its design, if it deviates from its original design, then it has to evaluate that change against the criteria in 10 CFR § 70.72 to determine if a license amendment is required or if it can make the change without NRC approval. If the change cannot be made without prior NRC approval, an amendment will be submitted and reviewed by the staff. If the change can be made without prior NRC approval, then the licensee can proceed with the change and inform the NRC of the change during the annual submission of facility changes and ISA Summary page changes.

Design differences are not expected to be identified by the Region Staff. As discussed above, the Region will work with USEC to identify those IROFS boundary packages needed prior to those IROFS being constructed. However, if differences are identified during these inspections, either the IROFS boundary packages could be changed to be consistent with the license requirements or USEC would need to evaluate the changes needed through the 10 CFR § 70.72 process and, if necessary, request an amendment.

If USEC were to proceed with construction prior to completion of the IROFS boundary packages, then it assumes the risk that, if differences are identified later, either physical changes to the facility will be required to return the facility to compliance or it will seek an amendment providing a justification for the difference in the design.

Q8: Explain what is meant by "samples of material prepared in a vertical slice fashion," and explain how this review approach is consistent with NUREG-1520 and NUREG-1513.

A8: (RW) The intent of these statements was to describe a review process which is prescribed by NUREG-1520 for the purpose of reviewing details in the development of safety and design information, the results of which are presented in the ISA Summary. NUREG-1520 recommends that the ISA review include an ISA methods review, a horizontal review, and a vertical slice review. Staff Exhibit 1 at A-30 - A-35. The ISA methods review is (1) to ensure that the applicant selected appropriate ISA method(s) for each facility process and (2) to ensure that they were correctly applied in conducting the ISA. The purpose of the horizontal review is to ensure completeness of the ISA. The vertical slice review examines how the ISA methods were applied to a selected subset of facility processes. The Staff performed four on-site reviews of ISA documentation in which it performed vertical slice reviews, ISA methods reviews, and horizontal reviews as documented in on-site review summaries (Memorandum from Yawar Faraz, NRC, to Joseph Giitter, NRC, "October 25-27, 2004, USEC, Inc. American Centrifuge Plant Integrated Safety Analysis Onsite Review," Dec. 9, 2004, Staff Exhibit 56; Memorandum from Yawar Faraz, NRC, to Joseph Giitter, NRC, "November 8-10, 2004, USEC, Inc. American Centrifuge Plant Integrated Safety Analysis Onsite Review," Dec. 15, 2004, Staff Exhibit 57; Memorandum from Yawar Faraz, NRC, to James Clifford, NRC, "August 15-17, 2005, USEC, Inc. American Centrifuge Plant Integrated Safety Analysis Onsite Review," Oct. 6, 2005, Staff Exhibit 58; Memorandum from Stan Echols, NRC, to Joseph Giitter, NRC, "April 2-4, 2006, On-Site Review Summary: Vertical Slice Review," May 23, 2006, Staff Exhibit 59) and summarized in Section A.3.2 of the Staff's SER.

For the ACP ISA methods review, the Staff evaluated the overall ISA methodology to assure that identified sequences were properly screened for credibility and that the ISA methodology selected by the applicant assures compliance with the 10 C.F.R. § 70.61 performance requirements. Special attention was given to mitigated sequences where the failure of multiple IROFS needs to be considered.

The horizontal slice review evaluated the completeness of the ISA by selecting various processes and evaluating the application of the ISA across that process, for example, the breakdown of accident sequences with several potential initiators. The horizontal review utilized the staff experience with similar facilities and processes to assure that all credible accident sequences were considered.

The vertical slice review involved the selection of a subset of facility processes and the risk informed selection of accident sequences within these processes. NUREG-1520 recommends that the staff perform a vertical slice review for 5 to 10 NCS significant processes, 1 to 3 fire significant processes, and 1 to 3 chemical/radiological/environmental-significant processes. The review included 10 criticality related sequences, 9 fire or explosion related sequences and 12 chemical/radiological/environmental-significant sequences involving over 6 process areas. The vertical slice review was used to examine the underpinnings of calculations, conclusions, and the design of safety programs that result from the ISA as well as safety information that is not identified in the ISA Summary

Q9: How close does this approach come to a "100 percent review"?

A9: (RW) These reviews did not comprise a 100 percent review. However, through these reviews the staff has verified, with reasonable assurance, that the applicant has performed an ISA adequate to identify and evaluate those hazards and potential accidents as required by the regulations.

Q10: Elaborate on the degree to which other regulations in 10 CFR Part 70 "apply to licensing review under Part 70" (NRC Staff Response at 30) and explain why the NRC Staff concluded that they "do not directly pertain to the required level of detail needed in performing a licensing review."

A10: (TJ) In the August 4, 2006, memorandum from R. Pierson (Staff Exhibit 60) and in response to Board Question S2-1, the NRC Staff cited specific regulatory citations that addressed the level of detail needed to be submitted in an application and needed to make a licensing decision for an application for a special nuclear materials license. In the September 13, 2006, memorandum from the two individuals (*see* Differing Professional Opinion, November 11, 2005, Staff Exhibit 62 at 2), the two individuals stated that the August 4, 2006, memorandum from R. Pierson (Staff Exhibit 60) contains "an incomplete list of the applicable regulation and does not provide a full picture of what is required for licensing."

In their first example, the two individuals refer to 10 CFR § 70.66(a). This requirement states as follows:

> "An application for a license from an applicant subject to subpart H will be approved if the Commission determines that the applicant has complied with the requirements of §§ 70.21, 70.22, 70.23, and 70.60 through 70.65."

Not every regulation cited in 10 CFR § 70.66(a) addresses the level of design detail required in an application. Only those sections that relate specifically to the required level of design detail were cited in the Pierson memorandum. For example, the regulations in 10 CFR § 70.21 provide general requirements for filing a application for a special nuclear materials license, but do not describe the level of detail needed to conduct a licensing review.

The regulations in 10 CFR § 70.22 provide general requirements for the content of an application. The only subsection of 10 CFR § 70.22 that is directly applicable to the level of detail issue is 10 CFR § 70.22(a)(7), which is cited in the August 4, 2006, memorandum and the Staff response to Board Question S2-1. The other sections provide general requirements for an application, but do not directly address the level of detail issue.

The regulations in 10 CFR § 70.23 provide general requirements for the approval of an application, but do not specifically address the level of detail needed for a licensing review.

The regulations in 10 CFR §§ 70.60 through 70.65 provide requirements for the applicability of subpart H to 10 CFR part 70 (10 CFR § 70.60); requirements for performance requirements (10 CFR § 70.61); requirements for the safety program and integrated safety analysis (10 CFR § 70.62); requirements for new facilities or new processes at existing facilities (10 CFR § 70.64); and requirements for additional content of applications (10 CFR § 70.65). Of these requirements, the only ones that directly apply to the issue of level of detail needed to be provided are 10 CFR §§ 70.65(b)(3) and 70.65(b)(6), which are cited in the August 4, 2006, memorandum and the Staff response to Board Question S2-1.

In their next set of examples, the dissenters two individuals refer to 10 CFR §§ 70.22(a)(8), 70.23(a)(3), and 70.23(a)(4). Staff Exhibit 62 at 3. The requirement in 10 CFR § 70.22(a)(8) states that an application shall contain:

> "Proposed procedures to protect health and minimize danger to life and property (such as procedures to avoid accidental criticality, procedures for personnel monitoring and waste disposal, post-criticality accident emergency procedures, etc.)"

The requirement in 10 CFR § 70.23(a)(3) states that one of the determinations the Commission must make is the following:

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

The requirement in 10 CFR § 70.23(a)(4) states that one of the determinations the Commission must make is the following:

> "The applicant's proposed procedures to protect health and to minimize danger to life or property are adequate."

These above requirements are broad requirements that do not provide specific insight into the level of detail required to make the determination. Therefore, these requirements were not specifically cited in the August 4, 2006, memorandum and the Staff response to Board Question S2-1.

In their next example, the two individuals refer to 10 CFR §§ 70.61(b), 70.61(c), 70.61(d), and 70.61(e) and suggest that these requirements should also have been included in the August 4, 2006, memorandum. These requirements provide general information on what must be evaluated in the integrated safety analysis and address the allowable risks for high-consequence, intermediate-consequence, and criticality events, and the requirements for items relied on for safety. Again, these specific regulations do not provide specific insight into the level of detail needed to make a licensing decision.

Lastly, in their September 13, 2006, memorandum, the two individuals refer to 10 CFR § 70.65(b)(4), which states that the integrated safety analysis summary must contain:

> "Information that demonstrates the licensee's compliance with the performance requirements of § 70.61, including a description of the management measures; the requirements for criticality monitoring and alarms in § 70.24 and, if applicable, the requirements of § 70.64."

This specific requirement does not provide specific insight into the level of detail needed to make the licensing decision, and was, therefore, not cited in the August 4, 2006, memorandum and the Staff response to Board Question S2-1.

In conclusion, Staff considers that the clear language of the regulations and the history of the development of Subpart H to 10 CFR Part 70 in SECY-00-0111 demonstrate that the intent of the regulations is not to require final design detail for the purpose of performing a licensing review. In addition, Staff, based on its evaluation in its Safety Evaluation Report (Staff Exhibit 1), considers that USEC has met all the applicable requirements in 10 CFR Parts 40 and 70 required for a uranium enrichment facility. Staff considers that USEC provided sufficient information, as required under the regulations, in the license application and integrated safety analysis summary so Staff could perform its licensing review and make its determinations as presented in the SER.

Q11: Does this conclude your testimony?

A11: Yes.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 16 2007.

Timothy Johnson

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 16 2007.

Jay Henson

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 16, 2007.

Brian W. Smith

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 16, 2007.

Rex Wescott