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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
GE-HITACHI GLOBAL LASER ENRICHMENT LLC	)	Docket No. 70-7016-ML
	)	
	)	ASLBP No. 10-901-03-ML-BD01
(GLE Commercial Facility)	)	
	)	

NRC STAFF TESTIMONY RELATED TO TOPIC 2:
LICENSING AN EVOLVING DESIGN

Q1: Please state your name, occupation, employer, and your professional qualifications.

A1: (TJ) My name is Timothy C. Johnson. I am a Senior Project Manager in the Nuclear Regulatory Commission's (NRC's), Office of Nuclear Material Safety and Safeguards (NMSS), Division of Fuel Cycle Safety and Safeguards (FCSS). A statement of my professional qualifications is attached. I have been the project manager for the GE-Hitachi Laser Enrichment LLC (GLE) Facility licensing since October 2006.

A1: (BS) My name is Brian W. Smith. I am the Chief of the Uranium Enrichment Branch in the Nuclear Regulatory Commission's (NRC's), Office of Nuclear Material Safety and Safeguards (NMSS), Division of Fuel Cycle Safety and Safeguards (FCSS). A statement of my professional qualifications is attached. As the Chief of the Uranium Enrichment Branch, I am responsible for providing oversight for the licensing reviews for the USEC Inc. Lead Cascade and American Centrifuge Plant, Louisiana Energy Services, and General Electric-Hitachi Global Laser Enrichment LLC. I have been the Chief of the Uranium Enrichment Branch since 2003.

Q2: Please describe your responsibilities with regard to the NRC Staff's (Staff) review of the license application (LA) for the proposed GE-Hitachi Global Laser Enrichment LLC (GLE) Facility in Wilmington, North Carolina.

A2: (TJ) As the Licensing Project Manager (PM) for the GLE project I oversaw the licensing review of the application and the preparation of NUREG-2120 , "Safety Evaluation Report for the General Electric-Hitachi Global Laser Enrichment LLC Laser-Based Uranium Enrichment Plant in Wilmington, North Carolina" (SER) (Ex. NRC001). In addition to overseeing the licensing review, I reviewed Chapters 1 and 2 of the LA, which relate to General Information and Organization and Administration. My review of those sections is documented in Chapters 1 and 2 of NUREG-2120 (Ex. NRC001).

A2: (BS) As Branch Chief, I was the first-line manager responsible for the oversight of the technical review performed on the license application and preparation of the SER (Ex. NRC001) that documents the safety review prepared by NRC Staff.

Q3: What is the purpose of your testimony today?

A3: (TJ, BS) To explain the Staff's approach to reviewing the design of the proposed GLE facility and the regulatory basis for that approach, focusing on (1) how the staff decides the level of design detail and finality needed to conduct a safety review and make a licensing decision; (2) significant aspects of the design of the proposed GLE facility that are still evolving and which could impact the safety of the facility; and (3) if a license is granted, how the staff will ensure that any future changes to the design of the proposed GLE facility will fall within the parameters of a license issued on the basis of the current design.

Q4: What regulatory requirements are applicable to the Staff's review of the design of the proposed GLE facility?

A4: (TJ, BS) The regulations applicable to the review of the design of the proposed GLE facility are found in 10 CFR Part 70. They include:

- 10 CFR 70.22(a)(2), which requires an applicant to provide information on the activity for which licensed material is requested or is produced, the place at which the activity is to be performed, and the general plan for carrying out the activity.

- 10 CFR 70.22(a)(7), which requires the applicant to provide information describing the equipment and facilities which will be used 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.)
- 10 CFR 70.23(a)(3), which states that an application for a license will be approved if the applicant's proposed equipment and facilities are adequate to protect health and to minimize danger to life or property.
- 10 CFR 70.61, which requires an applicant to evaluate, in an integrated safety analysis, its compliance with the performance requirements in this section.
- 10 CFR 70.62(a), which requires an applicant to establish and maintain a safety program that demonstrates compliance with 10 CFR 70.61.
- 10 CFR 70.62(b), which requires an applicant to maintain process safety information to enable the performance and maintenance of an integrated safety analysis. This process safety information must include information pertaining to the hazards of the materials used or produced in the process, information pertaining to the technology of the process, and information pertaining to the equipment in the process.
- 10 CFR 70.62(c), which requires an applicant to conduct and maintain an integrated safety analysis (ISA), that is of appropriate detail for the complexity of the process, that identifies the radiological hazards, chemical hazards, facility hazards, potential accident sequences, consequences and likelihood of potential accidents, and items relied on for safety identified pursuant to 10 CFR 70.61(e) to include the characteristics of its preventative, mitigative, or other safety function, and the assumptions and conditions under which the item is relied on to support compliance with the performance requirements in 10 CFR 70.61.

- 10 CFR 70.64(a), which requires an applicant for a new facility to meet baseline design criteria.
- 10 CFR 70.64(b), which requires that facility and system design and facility layout be based on defense-in-depth practices.
- 10 CFR 70.65(a) – requires that an application include a description of the applicant's safety program.
- 10 CFR 70.65(b), which requires the submittal of an integrated safety analysis summary, which is to contain a description of the process (defined as a single reasonably simple integrated unit operation within an overall production line) analyzed 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.
- 10 CFR 70.66, which states that an application for a license will be approved if the applicant has complied with the requirements in 10 CFR 70.21, 70.22, 70.23, and 70.60 through 70.65.
- 10 CFR 70.72, which requires a licensee to establish a configuration management system to evaluate, implement, and track changes to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel. This regulation also allows a licensee to make changes without prior NRC approval if new types of accident sequences that exceed the performance requirements in 10 CFR 70.61 are not created; the change does not involve a new process, technology, or control system for which the licensee has no prior experience; the change does not remove, without at least an equivalent replacement of the safety function, an item relied on for safety listed in the integrated safety analysis summary necessary for compliance with the performance requirements of 10 CFR 70.61; the change does not alter a sole

item relied on for safety; or the change is not prohibited by this section, license condition, or order.

Q5: Which guidance documents did the Staff use to evaluate the design of the proposed GLE facility?

A5: (TJ, BS) The Staff primarily used NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility" (SRP)(Ex. NRC005). Within NUREG-1520 are references to numerous other NRC and industry guidance documents.
Within NUREG-1520, facility design is addressed in Chapter 3, "Integrated Safety Analysis and Integrated Safety Analysis Summary," Chapter 5, "Nuclear Criticality Safety," Chapter 6, "Chemical Process Safety," and Chapter 7, "Fire Safety." As discussed in the response to Question 13 below, NUREG-1520 was revised to better address the level-of-detail issue. NUREG-1520, Revision 1, was issued in May 2010, after the GLE license application was submitted. Therefore, NUREG-1520, Revision 0, was used in this review. The staff, however, had knowledge of the proposed guidance in Revision 1 and the review of the GLE application was informed by this information.

The Staff also considered in its review the policy guidance in an August 4, 2006, memorandum to the Division of Fuel Cycle Safety and Safeguards staff (Pierson Memo) (Ex. NRC021). The Pierson Memo addresses the level of detail that an applicant has to provide when submitting an application for a license under 10 CFR Part 70 and that the Staff requires to make a licensing decision on such an application.

Q6: Please provide a brief explanation of the purpose and use of the SRP.

A6: (TJ, BS) The SRP (Ex. NRC005) provides generic guidance for reviewing and evaluating the health, safety, and environmental protection aspects of applications for licenses to possess and use special nuclear material in nuclear fuel cycle facilities. The principal purpose of the SRP is to ensure the quality and uniformity of reviews conducted by the Staff. Because the SRP describes the scope, level of detail, and acceptance criteria for reviews, it

also serves as regulatory guidance for applicants who need to determine what information to present in a license application. Because the SRP is a guidance document, the information presented in the SRP does not preclude licensees or applicants from suggesting alternative approaches to those specified in the SRP to demonstrate compliance with applicable regulations. Should a licensee or applicant suggest alternative approaches, the Staff retains the responsibility to make an independent determination concerning the adequacy of the applicant's proposed approaches. (Ex. NRC005 at xix-xxiii). The SRP was developed after extensive communication with fuel cycle licensees to ensure that all necessary safety and environmental issues were addressed.

Q7: Does the SRP apply to different types of fuel facilities?

A7: (TJ, BS) Yes. The SRP was developed as a generic document for licensing fuel cycle facilities under 10 CFR Part 70, including fuel fabrication facilities and uranium enrichment facilities like the GLE facility, the Louisiana Energy Services (LES) National Enrichment Facility (NEF) gas centrifuge facility, the AREVA Eagle Rock Enrichment Facility (EREF) gas centrifuge facility, and the USEC, Inc. American Centrifuge Plant (ACP) gas centrifuge enrichment facility. The Staff prepared a separate review plan for the Mixed Oxide (MOX) Fuel Fabrication Facility, NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" because of the significant differences in the MOX facility functions and hazards associated with the processing of weapons-grade plutonium.

While it is true that there are differences among fuel cycle facilities, hazards that will exist at the proposed GLE facility are similar to the types of hazards at other fuel cycle facilities for which NUREG-1520 was prepared. These hazards include handling of uranium hexafluoride cylinders, processing of uranium hexafluoride as a gas and sometimes as a liquid, use of autoclaves for feeding and sampling uranium, nuclear criticality, equipment decontamination operations, and laboratory activities.

Q8: How does the Staff adapt the generic SRP to review applications for different types of 10 CFR Part 70 facilities?

A8: (TJ, BS) The relative risk of the facility necessarily informs the Staff's review. The Staff's review of each type of license application (e.g., enrichment facility or fuel fabrication facility) will focus on the specific types of hazards associated with the particular technology. The goal of the reviews is to determine whether an adequate level of safety is provided to protect the health and safety of the public and the environment. Specific regulatory requirements for each type of license are found in the applicable sections of the NRC's regulations. The Staff recognizes that the types and magnitudes of potential hazards vary greatly between the various types of fuel cycle facilities and even within each type. Based on the processes performed at each type of facility, the proposed GLE facility has the lowest level of potential hazards, fuel fabrication facilities have the next level of hazard, and the MOX fuel fabrication facility has the highest level of hazard of all 10 CFR Part 70 fuel cycle facilities.

For fuel cycle facilities, excluding a spent fuel reprocessing plant or a MOX facility that processes plutonium, the main hazard associated with a loss of material confinement is chemical exposure. For those facilities that process low enriched uranium (i.e., uranium enrichment facilities and fuel fabrication facilities), the primary chemical hazards include soluble uranium compounds, which present a heavy metal toxicity concern, and hydrogen fluoride (HF), which is a product of the chemical reaction between UF_6 and water (moisture from the air). Qualitatively, the chemical risks posed by these facilities are far below those found at a typical chemical plant. The external radiological dose rates are minimal, and the chemical toxicological effects to individuals constitute the predominant hazard until about 18 percent U_{235} enrichment, at which point internal radiation dose becomes the primary hazard. Fuel fabrication facilities possessing enriched uranium with enrichments greater than 20 percent (Category I facilities) require consideration of both chemical and radiological hazards.

As a consequence of the above, the safety evaluations for each type of facility will vary as each is tailored to the relative risks involved. Thus, while the guidance in the SRP is applicable to a laser-based uranium enrichment facility, the Staff's review is informed by the fact that the overall risk of this type of facility is lower than that of other types of fuel facilities licensed by the NRC.

Q9: Discuss the staff's approach to reviewing the design of the proposed GLE Facility.

A9: (TJ, BS) The process for reviewing the LA for the proposed GLE Facility was the same process described in the introduction to NUREG-1520 (Ex. NRC005). In accordance with this process, GLE submitted licensing basis documents describing the proposed facility and the health and safety, environmental, and security programs it intends to implement. The Staff reviewed the licensing basis documents in accordance with the areas of review discussed in NUREG-1520 (Ex. NRC005), using the acceptance criteria for each of the review areas. When the Staff needed more information about a particular topic, the Staff prepared Requests for Additional Information (RAIs) and transmitted them to GLE. GLE provided responses and revised the licensing basis documents as needed. If additional clarifying information was needed, the Staff conducted conference calls and held meetings with GLE, and documented the results of those conference calls and meetings. In addition, the Staff conducted an on-site review of the calculations and supporting information that GLE prepared for the Integrated Safety Analysis (ISA). This review consisted of a vertical-slice review of the ISA where specific topics having the greatest potential for accident consequences were evaluated in more detail. The staff's review of the design of the facility was ultimately documented in the SER (NUREG-2120) (Ex. NRC001).

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. The focus on programmatic provisions is reflected in the regulatory requirements in 10 CFR 70.65(b)(3) and 70.65(b)(6) that call for "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." This is also reflected in the various chapters of the standard review plan, NUREG-1520 (Ex. NRC005). By "programmatic," we are referring to the reviews of programs described in NUREG-1520, such as the integrated safety analysis program, the nuclear criticalilty safety program, the radiation protection program, the fire and chemical safety programs, commitments to implementing consensus codes and standards, etc. Based on this understanding, the licensing review focuses on the applicant's programmatic commitments and, consequently, the licensing decision is ultimately based on a sufficient level of detail to understand process system functions and functionally how items relied on for safety can perform their intended safety function and be available and reliable to perform that function. The level of detail required for a licensing decision, therefore, does not require a final detailed facility design or an absolutely complete identification of all supporting items relied on for safety and accident sequences, or a review of the detailed implementing procedures. Instead, sufficient information has to be provided to understand the process and functions of items relied on for safety and to provide reasonable assurance that the integrated safety analysis summary is complete.

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

For uranium enrichment facilities, to ensure that the applicant's programs have been sufficiently implemented and commitments have been properly applied in the final facility design

and in the constructed facility, the regulations in 10 CFR 40.41(g) and 10 CFR 70.32(k) state that:

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

This requirement, applied through inspections, and not by licensing reviews, will ensure that the programmatic commitments made by the licensee are properly implemented in the as-built facility. During this inspection process, the NRC reviews the final design of the facility and the procedures and other measures that have been prepared to implement the licensee's commitments that are reflected in the license.

Q10: How do the regulatory requirements identified in Question 4 address the level of design detail required for reviewing a license application or making a licensing decision?

A10: (TJ, BS) The regulations in 10 CFR Part 70 provide general, but not detailed, requirements on the level of detail for a license application and an integrated safety analysis. Similarly, NUREG-1520 (Ex. NRC005) provides general guidance on the required level of detail. The regulations and the SRP were written in a general way to accommodate different types of facilities that may have differing levels of hazards. For example, a facility proposing possession of large quantities of plutonium and using complex operations would require a greater level of detail than a facility using relatively small quantities of uranium with low U-235 enrichment levels using relatively simple operations.

Not every regulation cited in 10 CFR Part 70 addresses the level of design detail required in an application. For example, the regulations in 10 CFR 70.21 provide general requirements for filing an application for a special nuclear materials license, but do not describe the level of detail needed to conduct a licensing review. Similarly, the regulations in 10 CFR 70.22 provide general requirements for the content of an application, but in most cases do not address level-of-detail. 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 states that each application shall contain:

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

The 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 requirements for approval of an application that are relevant to facility design are found in 10 CFR 70.23(a)(3) and (a)(4). These requirements state that an application will be approved upon a finding that the applicant is qualified, the proposed equipment and facilities are adequate to protect health and minimize danger to life or property, and that the proposed procedures are adequate. As a technical matter, it is for the Office of Nuclear Material Safety and Safeguards to determine how final the design must be to make this finding. All of the above requirements are broad requirements that do not provide specific insight into the level of detail required to make the determination.

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).

The regulations in 10 CFR 70.61 require each applicant to evaluate, in an integrated safety analysis performed in accordance with 10 CFR 70.62, compliance with the performance requirements in 10 CFR 70.61(b), 10 CFR 70.61(c), and 10 CFR 70.61(d). These requirements provide general information on what must be evaluated in the integrated safety analysis; address the allowable risks for high-consequence, intermediate-consequence, and criticality events; and identify the requirements for items relied on for safety. However, these regulations

do not provide specific insight into the level of design detail needed to make a licensing decision.

The regulations in 10 CFR 70.65 describe the requirements for the contents of an integrated safety analysis summary that is required to be submitted with the application. The requirements 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)(1)(i)-(iii) and a general description of the types of accident sequences.

The regulations in 10 CFR 70.65(b)(6) require that the integrated safety analysis summary contain:

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

As discussed in the response to Question 9 above, these requirements reflect a programmatic level of review. Based on the information in the LA and the integrated safety analysis summary provided as required under 10 CFR 70.65, licensing decisions are made as required under 10 CFR 70.21, 70.22, 70.23, and 70.60 through 70.66. These decisions include compliance with the performance requirements, the baseline design criteria, defense-in-depth, and the adequacy of management measures.

Q11: With respect to level of detail, how did the staff's approach to reviewing the design of the proposed GLE facility compare with the approach used in reviewing the designs of other recently-licensed gas centrifuge enrichment facilities?

A11: (TJ, BS) The staff used the same approach for reviewing the proposed GLE facility as it did for the Louisiana Energy Services (LES) National Enrichment Facility (NEF), the USEC, Inc.(USEC) American Centrifuge Plant (ACP), and the AREVA Eagle Rock Enrichment

Facility (EREF). Specifically, the staff applied a reasonable assurance standard and reviewed programmatic commitments to make the licensing decisions for the LES, USEC, and AREVA enrichment facilities. Therefore,the staff's approach to addressing the level of detail in the facility design for the proposed GLE facility was the same as the approach used for the other recently licensed gas centrifuge facilities.

The staff's reviews of the license applications for LES, USEC, and AREVA were subject to mandatory hearings (and in the case of LES, a contested hearing as well). In each case, the Atomic Safety and Licensing Board concluded that the staff had adequately performed its safety review.

Q12: Are there any significant differences between the proposed GLE facility and gas centrifuge facilities that would warrant a different level of design detail for the proposed GLE facility?

A12: (TJ, BS) There are no significant differences between the proposed GLE facility and the other gas centrifuge projects that would warrant a different level of design detail for the proposed GLE facility because the facility hazards at the GLE facility and the gas centrifuge plants are similar. The GLE facility differs from gas centrifuge facilities in that enrichment takes place using a process using laser energy rather than by a mechanical process. The facilities are similar in that there will be no licensed material other than natural uranium, depleted uranium, or uranium enriched up to 8 percent U_{235} present at the GLE facility except sealed sources used for instrument calibration. Similarly, licensed materials at the GLE facility and the gas centrifuge facilities are mostly contained in uranium hexafluoride cylinders or in the cascade vessels and piping, and open sources of uranium would be present only in the laboratories and in decontamination facilities. At the GLE facility or a gas centrifuge facility, the only significant chemical hazard is from uranium hexafluoride. In contrast, fuel fabrication facilities use other hazardous chemicals in their processes that may present exposure hazards to workers and the public.

Q13: Discuss the Differing Professional Opinion (DPO) filed in the USEC Inc. ACP licensing action.

A13: (TJ, BS) During the review of the license application for the USEC ACP, several NRC Staff members from the Office of Nuclear Materials Safety and Safeguards (NMSS) and the Office of Regulatory Research filed a Differing Professional Opinion (Ex. NRC022) expressing concerns with the guidance provided in the Pierson Memo (Ex. NRC021) regarding the level of detail required for licensing of facilities under 10 CFR Part 70. The DPO raised issues regarding whether the guidance in the memorandum complied with the regulatory requirements of 10 CFR Part 70 and the guidance in NUREG-1520 (Ex. NRC005).

The concerns raised in the DPO were addressed through the NRC's Differing Professional Opinion process. Decisions by the Director of the Office of Nuclear Material Safety and Safeguards, dated July 24, 2007 (Ex. NRC023), and the Executive Director of Operations, dated January 9, 2008 (Ex. NRC024), supported the Staff's use of the policy guidance in the Pierson Memo. In his review of the DPO, the Director of the Office of Nuclear Material Safety and Safeguards (NMSS), (Ex. NRC023) concluded that a programmatic review, as described in the Pierson Memo, is consistent with the requirements of Part 70. He also concluded that the SRP can be interpreted to allow a programmatic review (Ex. NRC023). After this decision was appealed to the Executive Director of Operations (EDO), the EDO also concluded that the actions taken by the staff in response to the DPO were adequate (Ex. NRC024). In their decisions, both the Director of NMSS and the EDO recommended clarifying the guidance on level-of-detail in the next revision to NUREG-1520. When Revision 1 to NUREG-1520 was issued in 2010, it included clarifying guidance in the Introduction and in Section 3.4.3 to address these recommendations (Ex. NRC020).

In view of the DPO decisions, Staff considers that the language of the regulations demonstrates that the intent of the regulations is not to require final design detail for the purpose of performing a licensing review. Final design detail for a facility is verified by inspection before

possession or material and operation. In addition, the Staff, based on its evaluation in the SER (Ex. NRC001), concluded that GLE met all the applicable requirements in 10 CFR Parts 40 and 70 required for a uranium enrichment facility. GLE provided sufficient information, as required under the regulations, in the license application and integrated safety analysis summary so the Staff could perform its licensing review and make its determinations as presented in the SER.

Q14: Are there aspects of the GLE design that are still evolving that could affect safety?

A14: (TJ, BS) Yes. The NRC staff understands that GLE is continuing to conduct design testing and studies on the laser-based enrichment process. These testing and design studies are expected to continue in the nearfuture. The primary developmental area that could affect safety is the final design of the separators, where GLE efforts are focused on optimizing uranium separation and material handling. GLE is also furthering the designs of the remainder of the facility, including the feed, withdrawal, blending, and sampling systems. However, the baseline designs for the remainder of the plant, as described in the Integrated Safety Analysis summary (Ex. GLE010), are not expected to undergo significant changes.

Q15: Which of the evolving design efforts are the most important from a safety perspective?

A15: (TJ, BS) NRC staff expects that the principal hazards in the facility will be the same as in other licensed uranium enrichment plants. That is, the most important facility hazard is in the liquid sampling system, where product cylinders are heated to conditions that cause the uranium hexafluoride to become a liquid so that it is possible to obtain representative samples of the product. NRC staff does not expect that GLE will make significant changes to the design of the liquid sampling system. Although the design of the separators is expected to change the most, the changes are not expected to significantly affect safety, because of the separator operating conditions and lack of moderation (see the response to Question 14 in the Staff's prefiled testimony on Topic 1A, Criticality Safety, for additional information).

Q16: If a license is granted to GLE, how will the staff ensure that any future changes fall within the parameters of the license issued on the basis of the current design?

A16: (TJ, BS) As a licensee, it will be GLE's responsibility 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., baseline design). As GLE completes its design, any deviations from its original design would have to be evaluated against the criteria in 10 CFR. 70.72 to determine if a license amendment is required or if GLE could make the change without NRC approval. Under 10 CFR 70.72(a), fuel cycle facility licensees must establish a configuration management system to evaluate, implement, and track each change to the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel. Licensees may make changes to these aspects of the facility without prior approval of the NRC unless the changes fall within the criteria in 10 CFR 70.72(c). Examples of changes that fall within the 10 CFR 70.72(c) criteria (and therefore requiring a license amendment) include changes that create new types of accident sequences that would exceed the 10 CFR 70.61 performance requirements unless mitigated or prevented; use of new processes, technologies, or control systems for which the applicant has no prior experience; changes that remove, without at least an equivalent replacement of the safety function, an item relied on for safety that is listed in the integrated safety analysis summary and is necessary for compliance with the performance requirements of 10 CFR 70.61; and changes that remove or alter an item relied on for safety, listed in the integrated safety analysis summary, that is the sole item preventing or mitigating an accident sequence that exceed the performance requirements in 10 CFR Part 70.61. Changes that fall outside of those criteria do not require NRC approval unless they are otherwise prohibited by license condition or order (*see* 10 CFR 70.72(c)(4)). Therefore, under the configuration management program and the implementation of the change process in 10 CFR 70.72, the licensee would evaluate all design changes to ensure that the performance requirements in 10 CFR 70.61 would continue to be met. In

addition, the Staff would review the annual update of the ISA Summary required to be submitted under 10 CFR 70.72. The Staff would also review any changes made to the LA in accordance the change process addressed in Section 1.2.3.7.2 of the SER.

Before the proposed GLE facility could begin operations, the NRC is required, in accordance with 10 CFR 40.41(g) and 70.32(k), to verify through inspection that the facility has been constructed in accordance with the requirements of the license. As a part of this inspection effort, the NRC must verify that the final designs of, safety significant structural features, equipment, and components, including any design changes, 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. It is expected that the construction inspection will involve multiple inspections that will take place over the course of the construction effort. Because GLE is planning to construct the facility in phases, separate determinations will be required by 10 CFR 40.41(g) and 70.32(k) for each phase before operations in that phase can begin.

In addition to the construction inspections required by 10 CFR 40.41(g) and 70.32(k), NRC will also perform an Operational Readiness Review (ORR) to ensure that other safety programs, such as the radiation protection program, nuclear criticality safety program, chemical and fire safety programs, and emergency preparedness programs, are in-place prior to operations and are consistent with the applicant's commitments in the license. Because the language in 10 CFR. 40.41(g) and 70.32(k) explicitly addresses only construction, Staff considers the construction inspections to be distinct from the other inspections performed in the ORR. However, the Staff considers the construction inspections to be a part of the overall ORR.

NRC Region II staff, with assistance from NMSS and contractor staff, will inspect the construction of the facility on a risk-informed sample basis. 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. On a prioritized sample basis, 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. This prioritization will be based of the level of hazard, importance to safety, and significant changes that may have taken place from the baseline design.

The NRC's 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 ORR prior to plant operation.

GLE will be required to follow its procedure for defining the boundaries of each of its IROFS. The IROFS boundaries are documented in IROFS boundary packages and are required to be available for inspection at the time of the construction inspections and operational readiness reviews. These IROFS boundary packages, as defined in GLE's Quality Assurance Program Description (Ex. GLE011 at viii), are:

> [D]ocuments that contain the physical descriptions and parameters of structures, systems, and components that are used to meet the performance requirements of 10 CFR 70.61. IROFS boundary definition packages are also prepared for administrative procedures or worker actions, which are defined as IROFS. The boundary packages also identify the specific functions to be performed by an

> IROFS and identify any items that may affect the function of the IROFS. The boundary packages also identify the facility areas in which the IROFS is used, design and functional attributes, management measures, any open items, and supporting documentation (e.g., P&IDs [piping and instrumentation diagrams], schematics, etc.).

Therefore, IROFS boundary packages are integrated documents containing all relevant information pertaining to individual IROFS.

Although the operational readiness reviews will be conducted close to the time of plant operation, Region II staff will be conducting construction inspections throughout 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 GLE on the availability of certain IROFS boundary packages such that they can be inspected prior to inspection 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 packages.

Q17: Has LES or USEC Inc. made any significant changes in the design of their facilities, and if so, were amendments required?

A17: (BS) Both licensees have made significant changes in the design of their plants. LES has switched to a new model of centrifuge to perform the enrichment of the uranium. The change was requested in a license amendment request dated September 22, 2011 (Ex. NRC025). NRC Staff approved the amendment request on May 22, 2012 (Ex. NRC026). LES evaluated this facility change through its10 CFR 70.72 facility change process and determined that the change could be made without the submittal of an amendment request. However, an amendment was required to modify a table in the LES license application criticality safety chapter regarding control mechanisms and safety parameters (i.e., safe dimensions of

components) to prevent criticality in a individual centrifuge. Had the newer model centrifuge's dimensions been the same as the original model, no amendment request would have been required.

USEC submitted an amendment request on November 17, 2008, seeking approval to make changes to the feed and withdraw system design for the ACP (Ex. NRC027). USEC proposed revisions to the previously approved Integrated Safety Analysis (ISA) Summary and proposed change pages to the previously approved license application. The licensee submitted a detailed description and justification of the proposed changes which focused on consolidating feed, withdrawal, blending, and transfer capabilities in the X-3346 Feed and Withdrawal Building by eliminating the X-3356 Product and Tails Withdrawal Building and moving sampling capability to a new facility designated as the X-3344 Customer Services Building. Additionally, the licensee proposed removing the X-7756S, X-7746N, and X-7746E Cylinder Storage Yards. In making these changes, USEC simplified the design of their plant and significantly reduced the amount of liquid UF_6 present at the site, thereby reducing the overall risk of the facility. USEC evaluated this facility change in their 10 CFR 70.72 facility change process and determined that the change required a license amendment request to be submitted. NRC Staff approved the USEC amendment request on July 14, 2009 (Ex. NRC028).

Q18: Does this conclude your testimony?

A18: (TJ, BS) Yes.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket No. 70-7016-ML
	)	
GE-HITACHI GLOBAL LASER ENRICHMENT LLC	)	ASLBP No. 10-901-03-ML-BD01
	)	
	)	June 18, 2012
(GLE Commercial Facility)	)	

AFFIDAVIT OF TIMOTHY C. JOHNSON

I, Timothy C. Johnson, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Timothy C. Johnson

Executed at Rockville, Maryland
this 18th day of June, 2012

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of GE-HITACHI GLOBAL LASER ENRICHMENT LLC (GLE Commercial Facility)	))))))	Docket No. 70-7016-ML ASLBP No. 10-901-03-ML-BD01 June 18, 2012

AFFIDAVIT OF BRIAN W. SMITH

I, Brian W. Smith, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Brian W. Smith

Executed at Rockville, Maryland
this 18th day of June, 2012

Timothy C. Johnson
Statement of Professional Qualifications

CURRENT POSITION

Senior Project Manager
Uranium Enrichment Branch
Division of Fuel Cycle Safety and Safeguards
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C.

EDUCATION

M.S., Nuclear Engineering, Ohio State University
B.S., Mechanical Engineering, Worcester Polytechnic Institute

PROFESSIONAL

American Nuclear Society, Member
American Society of Mechanical Engineers, Member
American Society for Testing and Materials, Member

QUALIFICATIONS

Mr. Johnson is a nuclear engineer with over 39 years of work experience in industry and in the Federal government. He has been working in the areas of radioactive waste processing, low-level radioactive waste management, high-level radioactive waste management, decommissioning, and uranium enrichment facility licensing.
Mr. Johnson is currently the Licensing Project Manager of the General Electric-Hitachi Global Laser Enrichment (GE) uranium enrichment plant project in the Uranium Enrichment Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission. Mr. Johnson received a Bachelor of Science degree in Mechanical Engineering from Worcester Polytechnic Institute in Worcester, Massachusetts, in 1971 and a Master of Science degree in Nuclear Engineering from Ohio State University, in Columbus, Ohio, in 1973. Courses he has taken that are pertinent to his present discipline are in the areas of advanced mathematics, engineering design, mass and heat transport, thermodynamics, reactor theory, nuclear physics, nuclear power plant engineering, and health physics. He was elected to membership in Pi Mu Epsilon, the mathematics honorary society.

From January 1973 to August 1977, Mr. Johnson was employed by Stone & Webster Engineering Corporation in Boston, Massachusetts. As the offgas and ventilation filter system specialist, he was responsible for the technical adequacy of offgas and ventilation filter systems for pressurized water reactor, boiling water reactor, high temperature gas cooled reactor, and liquid metal fast breeder reactor projects. His responsibilities included ensuring that equipment met both applicable regulatory and equipment code requirements. He prepared master specifications for offgas and ventilation filter systems for use by project staff. He reviewed project specifications and performed technical reviews of vendor proposals. He also reviewed vendor procedures for qualification and testing of offgas and ventilation system components.

Since September 1977, Mr. Johnson has been employed by the U.S. Nuclear Regulatory Commission in the areas of radioactive waste management, decommissioning, and fuel cycle facility licensing.

From September 1977 to April 1984, Mr. Johnson had lead responsibility for the waste form performance aspects of low-level radioactive wastes to include radwaste processing, solidification, high integrity containers, and volume reduction systems. In this capacity, he developed programs for analyzing, evaluating, coordinating, and recommending licensing actions related to the waste form and waste classification areas of 10 CFR Part 61. These responsibilities have specifically included coordinating the development of the waste form and waste classification requirements and preparing the appropriate sections for: (1) the low-level waste management regulation, 10 CFR Part 61; (2) the draft and final environmental impact statements that support 10 CFR Part 61; and (3) the technical positions on waste form and waste classification that provide guidance to waste generators for complying with the 10 CFR Part 61 requirements. He also acted as lead for an intra-agency task group for implementation for the 10 CFR Part 61 requirements at nuclear power plants.

During this time, Mr. Johnson also participated on a Task Force responsible for Three Mile Island Unit 2 (TMI-2) waste disposal issue resolution to include the evaluation of EPICOR-II, Submerged Demineralizer System, and decontamination solution wastes. He also prepared and coordinated the waste disposal section for the TMI-2 Programmatic Environmental Impact Statement.

From April 1984 to April 1987, Mr. Johnson was Section Leader of the Materials Engineering Section in the Division of Waste Management. In this capacity, he supervised a section that performed technical and engineering evaluations of low-level and high-level radioactive waste packages. This included planning and executing section programs, providing technical direction and integration of materials concerns into NRC low-level and high-level waste licensing activities, and supervising the management of technical assistance programs.

From April 1987 to May 1992, Mr. Johnson was Section Leader of the Special Projects Section in the Division of Waste Management. In this capacity, he supervised a section responsible for mixed wastes, decommissioning of materials licensee facilities and power reactors, financial assurance for decommissioning materials licensees and low-level waste disposal facilities, greater than Class C wastes, low-level waste disposal site quality assurance, and the low-level waste data base.

From May 1992 to November 1999, Mr. Johnson was Section Chief of decommissioning sections in the Division of Waste Management responsible for developing and executing the Site Decommissioning Management Plan (SDMP), an agency effort to ensure that 17 decommissioning policy issues were resolved and over 40 non-routine decommissioning sites would be properly decommissioned. During this time, he acted as Project Manager for the decommissioning of the Chemetron site in Cleveland, Ohio, a controversial contaminated site located in a residential neighborhood. The site was remediated and the license terminated in 1998.

From November 1999 to the present, Mr. Johnson was a Senior Project Manager in the Division of Fuel Cycle Safety and Safeguards. In this position, he acted as deputy project manager for the Mixed Oxide Fuel Fabrication Facility licensing and project manager for the licensing of gas centrifuge uranium enrichment facilities. He also was the Project Manager of the Louisiana

Energy Services uranium enrichment plant from the project's inception in 2000, through licensing, and into initial plant construction until 2009. As Project Manager, he coordinated the licensing review of the licensing application. He also provided testimony in contested and mandatory hearings for this facility in the areas of uranium enrichment facility licensing, the disposition of depleted uranium, and decommissioning financial assurance. He is currently the Project Manager for the General Electric-Hitachi Global Laser Enrichment uranium enrichment plant responsible for coordinating the licensing review of the facility.

Mr. Johnson has prepared over 25 presentations and papers given at conferences and has been an instructor at American Society of Mechanical Engineering, Harvard School of Public Health, and NRC training courses in the areas of low-level waste management, decommissioning, and uranium enrichment facility licensing.

Brian W. Smith
Statement of Professional Qualifications

CURRENT POSITION

Chief, Uranium Enrichment Branch
Division of Fuel Cycle Safety and Safeguards
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C

EDUCATION

B.S., Nuclear Engineering, North Carolina State University

PROFESSIONAL

None

QUALIFICATIONS

Mr. Smith joined the NRC in 1996 and has been the Chief of the Uranium Enrichment Branch since September 2003. During this time, he has overseen the licensing reviews and ultimate license issuance for the USEC Inc. Lead Cascade, USEC Inc. American Centrifuge Plant, and Louisiana Energy Services National Enrichment Facility as well as the current licensing review for the GE-Hitachi laser enrichment plant. For two years, he also had the responsibility for the licensing actions associated with the two United States Enrichment Corporation gaseous diffusion uranium enrichment plants and the Honeywell uranium conversion plant. In his first year in this position, he also had the licensing responsibility for the Duke Cogema Stone and Webster construction authorization request for a mixed oxide fuel fabrication facility. He is responsible for the development of the budgets for each of these projects and obtaining the staff with the appropriate technical expertise to perform these reviews.

Prior to his current position, for a year and a half Mr. Smith served as a Senior Assistant in the Office of the Executive Director for Operations (OEDO). In this position, he functioned as OEDO liaison with NRC Program Offices, Commission staff, Regional Offices, and Commissioner's technical assistants; identified major issues requiring senior management's attention to improve office effectiveness in completing assigned work; reviewed Program Office Commission papers and correspondence; and assisted in the preparation of reports, correspondence, testimony, and public remarks.

Prior to moving to OEDO, Mr. Smith spent nearly six years in two positions within the Office of Nuclear Material Safety and Safeguards (NMSS). As an NMSS Regional Program Coordinator (Regional Coordinator), he performed several primary duties: 1) communicated with the Regions concerning materials events and other significant inspection and enforcement issues and briefed the NMSS/IMNS Division Director on a daily basis; 2) represented NMSS in the EDO's daily event briefings; and 3) performed enforcement activities, which include representing NMSS in enforcement panels with the Office of Enforcement, reviewing final enforcement packages, recommending concurrence by the IMNS Division Director, participating as the NMSS representative in Regional enforcement conferences with licensees, as necessary, and serving as the NMSS enforcement coordinator. As a Health Physicist, his duties involved participating in

event response functions, performing technical reviews, and drafting various types of documents, such as Information Notices, Policy and Guidance Directives, and a Commission Memorandum.

Prior to joining NRC, Mr. Smith served as an engineer for Program Management, Inc. for four years. His primary responsibility was to provide technical support in the development of occupational radiation protection policies, standards, and guidance for the Department of Energy, performed under a prime contract with DOE EH-52, Office of Worker Protection Programs and Hazards Management.