February 24, 2006

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LOUISIANA ENERGY SERVICES, L.P.	)	Docket No. 70-3103
	)	
(National Enrichment Facility)	)	ASLBP No. 04-826-01-ML
	)	

NRC STAFF PRE-FILED MANDATORY HEARING
TESTIMONY CONCERNING CRITICALITY

Q.1. Please state your name, occupation, by whom you are employed and your professional qualifications.

A.1. (WT) My name is William Troskoski. I am a Senior Technical Reviewer 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.

A.1. (HF) Harry Felsher, Nuclear Process Engineer, NRC, NMSS, FCSS. A statement of my professional qualifications is attached.

A.1. (KM) Kevin Morrissey, Nuclear Process Engineer, NRC, NMSS, FCSS. A statement of my professional qualifications is attached.

Q.2. Please describe your responsibilities with regard to the preparation of the Safety Evaluation Report (SER) for the National Enrichment Facility (NEF) in Lea County, New Mexico.

A.2. (WT) I was the primary reviewer of the applicant's Integrated Safety Analysis (ISA) and ISA Summary. My analysis of the applicant's ISA and ISA Summary is documented in Chapter 3.0 of the SER (see NUREG-1827). I was also the lead reviewer for chemical safety.

A.2. (HF) I was the reviewer of the applicant's nuclear criticality safety (NCS) information. My analysis of the applicant's NCS information is documented in Chapter 5.0 of the SER (see NUREG-1827).

A.2. (KM) I was assigned to provide technical assistance for the LES ISA Summary review and to provide detailed knowledge of the LES processes.

Q.3. What is the purpose of your testimony?

A.3. (WT, HF, KM) To explain the Staff's review of the ISA Summary submitted by the applicant and the NCS information described in the application and to address the Board's questions relating to Items Relied on for Safety (IROFS) and NCS.

Criticality Concepts

Q.4. Please describe the concept of criticality.

A.4. (WT, HF, KM) Criticality is the attainment of a self-sustaining nuclear chain reaction. The chain reaction occurs as atoms of a fissile material absorb slow neutrons and split (fission) into new lighter atoms (fission products) and additional neutrons that, in turn, interact with additional fissile atoms. When this process becomes self sustaining, meaning that it continues on its own, the process is said to be critical. The rate of fission and the associated production of neutrons is offset by the rate at which neutrons are lost to the system due to being captured or absorbed and the rate at which neutrons leak from the system due to the geometry of the system. Neutrons born from fission have high energy (fast neutrons) and in systems with low enriched uranium, such as the NEF, must be slowed down (thermalized) to cause additional fissioning of the material. Generally, water is used as the means to slow down, or moderate, neutrons to energies capable of causing fission.

Q.5. Please explain the conditions needed to achieve criticality and how to limit or control those conditions?

A.5. (WT, HF, KM) The conditions that contribute to achieving criticality for a low enriched uranium (LEU) system, like the system to be employed at the proposed NEF, are having enough nuclear material, having a non-favorable geometry, and having sufficient moderation.

The production rate for neutrons depends on the amount and type of fissionable material present in a system. Thus, limiting or removing fissile material (containing nuclides that can be fissioned by neutrons of any energy) is generally most significant in achieving subcriticality. Absorption processes remove neutrons that would otherwise participate in the fission chain reaction. The absorption process can be used to ensure subcriticality. Absorption can be increased by adding non-fissile materials. Neutron leakage also removes neutrons that would otherwise be part of the fission chain reaction. Neutron leakage is dependent on system geometry and density. For example, if geometry of a given composition and quantity of material is changed by increasing surface area, this will decrease density of the material and increase neutron leakage. On the other hand, neutron reflectors, such as graphite or concrete, decrease leakage by scattering back neutrons that would otherwise have been lost. Thus, limits on dimensions, densities and reflection are important to controlling leakage and achieving subcriticality. Controlling leakage by geometry is an important element in NCS. Generally, a situation where a container or piece of equipment cannot hold enough fissionable material to produce a criticality regardless of enrichment, concentration, reflection, or any other condition, is referred to as “subcritical by safe geometry.” Generally, a situation where a container or piece of equipment cannot hold enough fissionable material to produce a criticality based solely on enrichment, is referred to as “subcritical by favorable geometry.” Nuclear reactions are highly dependent on neutron energy. Fast neutrons are not readily captured in U^{235}, which is the fissile material in enriched uranium. Thus, the neutrons must lose energy and slow down or become “thermalized” in order to be readily captured and cause fission. The process by which

fast neutrons are slowed down is called moderation. The presence of a light element (such as hydrogen) is an effective moderator and is an important factor in achieving criticality.

Q.6. How is criticality calculated?

A.6. (WT, HF, KM) Criticality is calculated as the ratio of the production of neutrons to the destruction (loss) of neutrons. This ratio is expressed as the effective multiplication factor or k-effective (k_{eff}). A k_{eff} of 1.0 represents a system that is critical with an equal rate of neutron production and loss. When neutron loss exceeds neutron production, the system cannot sustain a nuclear chain reaction. The resulting k_{eff} is less than 1.0 and the system is called subcritical. When neutron production exceeds neutron loss, the resulting k_{eff} is greater than 1.0 and the system is called supercritical.

Q.7. How is the k_{eff} for a given system determined?

A.7. (WT, HF, KM) Experimental data provides valuable information on whether processes will become critical. However, the validity of comparing experimental results to plant conditions that are being evaluated depends on the extent to which the experimental arrangements match the process conditions being postulated. Because actual experimental data cannot be obtained for each potential design, computer codes have been developed to model the neutronic processes that occur in a system. The type of computer code used by the applicant is the Monte Carlo computer code (MONK 8A). This code models neutrons as individual particles which interact with nuclei randomly while obeying fundamental laws of probability under parameters that represent the conditions relevant to neutron behavior given the system modeled. The Monte Carlo code compares the number of neutrons generated to those at the beginning of the model to calculate a k_{eff} value with an uncertainty due to the random numbers being used in the Monte Carlo code.

Q.8. Is the facility that is the subject of this application designed to achieve criticality?

A.8. (WT, HF, KM) No. The processes involved at the proposed NEF and at other

fuel cycle facilities are designed and maintained to be subcritical. Criticality would only occur inadvertently.

Q.9. How are criticality accidents prevented?

A.9. (WT, HF, KM) There are a wide variety of controls used by fuel facility licensees to prevent an accidental criticality. These controls include passive and active engineering, as well as enhanced (augmented) and simple administrative controls. Passive-engineered controls are the preferred type of control because they use only fixed physical design features and do not rely on computers or human actions. Examples of these controls include a double roof to prevent water intrusion or a fixed storage rack that only physically allows a limited amount of nuclear material in a limited container size. Active-engineered controls are physical devices that monitor processes and respond to process deviations or upsets without human actions. Examples of active-engineered controls include a gamma monitoring device used to detect nuclear material in unwanted locations and to automatically close valves, or a level-sensor that monitors water level and closes a valve when a certain level is exceeded. Enhanced-administrative controls exist where a physical device and a human action constitute the control. Examples of these controls include a light on a console that alerts an operator to close a valve or an alarm that sounds in order to remind an operator to flip a switch. Simple-administrative controls exists when a human being performs an action based on that person's knowledge of a procedure. Examples of these controls include following a procedure to put only one item in a glovebox or following a procedure to pick the correct container to store nuclear material.

Regulatory Requirements

Q.10. Please explain the regulatory requirements in 10 C.F.R. Part 70, Subpart H that relate to nuclear criticality safety (NCS).

A.10. (WT, HF, KM) 10 C.F.R. Part 70, Subpart H, "Additional Requirements for Certain Licensees Authorized to Possess a Critical Mass Of Special Nuclear Material" apply to applicants, such as LES, that request authorization to possess greater than a critical mass of special nuclear material to engage in uranium enrichment processing. These regulations contain three separate requirements regarding NCS:

- Section 70.61(a) requires an applicant to evaluate, in the integrated safety analysis, its compliance with the performance requirements in § 70.61(b) and (c) to reduce the risk of events that could have significant impacts to workers or the public. Specifically, § 70.61(b) requires high consequence events to be highly unlikely and § 70.61(c) requires intermediate consequence events to be unlikely.
- Section 70.61(a) also requires compliance with § 70.61(d) which requires that nuclear criticality accidents be limited by assuring that under normal and credible abnormal conditions all nuclear processes are subcritical, including the use of an approved margin of subcriticality. Section 70.61(d) also requires that prevention, rather than mitigation, be the primary means of protection against an inadvertent criticality. The purpose of this requirement is to preclude a situation when an inadvertent criticality would be permitted so long as the dose thresholds of § 70.61(b) and 70.61(c) are not exceeded.
- Section 70.64(a)(9) requires that the design of new facilities and processes provide for criticality control including adherence to the double contingency principle. The double contingency principle means that process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Under 10 C.F.R. § 70.65(b)(4), the applicant is required to provide information that

demonstrates compliance with the performance requirements in § 70.61 in the integrated safety analysis summary. LES provided the required documentation in the National Enrichment Facility Integrated Safety Analysis Summary, Staff Exhibit 58-M.

Q.11. Are these three regulatory provisions consistent?

A.11. (WT, HF, KM) Yes, however, there has been some confusion about how to satisfy these requirements with a single analysis. Accordingly, the Staff developed guidance to clarify the relationship between these requirements in FCSS-Interim Staff Guidance (ISG)-03, Revision 0, "Nuclear Criticality Safety Performance Requirements and Double Contingency Principle," dated February 17, 2005, Staff Exhibit 59-M. As noted in that guidance, 10 C.F.R. § 70.61(b) and (c) are risk-informed and performance-based requirements, requiring that the overall risk of an accident, based on likelihood and potential consequences, be limited. However, application of these provisions alone would permit a facility to have an inadvertent criticality, provided that the consequences were low enough to meet the specified criteria. Accordingly, the more prescriptive provision of § 70.61(d) was included to ensure that all processes are designed to remain subcritical under normal and credible abnormal conditions.

Q.12. Is this consistent with the guidance in the Standard Review Plan, NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility" (SRP), Staff Exhibit 49-M?

A.12. (WT, HF, KM) Yes. Chapter 3.0 of the SRP discusses the content of the ISA Summary that is required under 10 C.F.R. § 70.65 and, under subsection (b)(4), must include information that demonstrates compliance with the performance requirements of § 70.61. Chapter 3.0 outlines a process by which the applicant can demonstrate compliance with § 70.61(b) and (c) by demonstrating that all potential high-consequence events are highly-unlikely and all potential intermediate-consequence events are unlikely. In general terms, the process requires the applicant to identify and assess all potential accidents as well as identify

controls for preventing or mitigating the consequences. These controls are referred to as Items Relied on for Safety (IROFS). Chapter 5.0 contains guidance on compliance with § 70.61(d) in section 5.4.3.4.4. For compliance with that provision, the guidance provides that an applicant's commitment to follow the regulatory requirements should be considered acceptable provided that the applicant commits, among other things, to use appropriate controls, to utilize appropriate standards and subcritical limits, and to implement a program that ensures double contingency protection when practicable.

LES Application

Q.13. What approach did LES use to demonstrate compliance with § 70.61?

A.13. (WT, HF, KM) LES combined the approach in Chapter 3.0 of the SRP for identifying IROFS with a safe-by-design approach for some aspects of NCS in order to comply with § 70.61(b). LES used the approach in Chapter 5.0 of the SRP to develop an NCS program, including a commitment to apply the double contingency principle in order to comply with § 70.61(d). LES documented the approach in its demonstration of compliance with § 70.61(b) in the Integrated Safety Analysis Summary, Staff Exhibit 58-M, submitted in accordance with §70.65(4).

Q.14. Could you please explain these elements, beginning with the safe-by-design approach?

A.14. (WT, HF, KM) Yes. LES proposed the use of a safe-by-design ISA method for those components related to NCS for which the only possible means of failure would be to incorrectly alter the component by replacement or physical alteration. LES proposed the following process, which was approved by the Staff, to demonstrate safe-by-design: Safe-by-design components are those components that are demonstrably safe by their physical size or arrangement and have been quantitatively determined to be safe. The quantitative analysis is

accomplished by means of criticality assessments. For components that are safe-by-volume, safe-by-diameter, or safe-by-slab thickness (favorable geometry components) LES demonstrated that the parameter values were less than those of a set of generic, conservative values for criticality from NRC-approved sources.

For the remaining safe-by-design components, LES performed detailed analysis and calculations to demonstrate an approved safety margin for NCS (defined as 10% between the actual parameter value of the component and the design value of the critical attribute). If the components meet the definition of safe-by-design, then the failure of the components will be highly unlikely and § 70.61(b) will be met. All analyses demonstrating that the definition of safe-by-design was met are in the NCS safety basis information. This safety basis information is used in the development of the ISA, which, in turn, is used to develop the ISA Summary. All safe-by-design components are considered items that may affect IROFS. As a result, Quality Level 1 requirements (the same requirements that apply to IROFS) apply to these safe-by-design components. The configuration management program required by § 70.72 will ensure the maintenance of the safety function of these safe-by-design components.

Q.15. What process did LES follow with regard to components which were not designated as safe-by-design?

A.15. (WT, HF, KM) LES used the approach outlined in Chapter 3.0 of the SRP. LES identified:

- The radiological hazards related to possessing or processing licensed material at its facility
- The chemical hazards of licensed material and hazardous chemicals produced from licensed material
- The facility hazards that could affect the safety of licensed materials and thus present an increased radiological risk by conducting a hazard analysis

- The potential accident sequences caused by process deviations or other events internal to the facility as well as credible external events
- The consequences and likelihood of occurrences of each potential accident sequence and the methods used to determine consequences and likelihoods
- The IROFS for each accident sequence and the characteristics of its preventive safety function.

LES identified potential hazards and accidents by means of a hazards analysis by a team composed of individuals with diverse technical disciplines and led by an individual qualified in the chosen hazard analysis technique using the HAZOP method. This method comes from the chemical industry and is a structured technique well suited to analyze processes during or after a detailed design stage. The HAZOP method is acceptable for identification of potential radiological, chemical and other facility hazards (e.g., fire, criticality), and potential accident sequences caused by process deviations or other events internal to the facility and credible external events, including natural phenomena that could lead to a loss of UF_6 confinement or an inadvertent criticality.

In assessing the risk associated with postulated accidents, LES assumed that every inadvertent criticality accident would have high consequences. Additionally, LES used only preventive IROFS for all criticality accidents. The results of this analysis are presented in the ISA and summarized in the ISA Summary. The ISA Summary includes a description of all accident sequences and any factors that prevent or mitigate the accident (IROFS), and the management measures that allow the IROFS to be available and reliable to perform their intended function when needed. LES included an accident sequence which is initiated by a ‘loss-of-safe-by-design attribute’ to account for the safe-by-design components for NCS. The likelihood of this accident was demonstrated to be highly unlikely by the safe-by-design process described above.

Q.16. How did LES present the information showing compliance with the performance requirements of § 70.61(b) and (c) in the ISA Summary?

A.16. (WT) The information is set forth in a risk matrix found in Table 3.1-6 of the ISA Summary.

Q.17. Please describe the risk matrix.

A.17. (WT) In order to satisfy the regulatory performance requirements, LES was required to evaluate the risk of accidents (i.e., likelihood x consequence). LES chose to display the three categories of consequence and likelihood as a 3 x 3 risk index matrix (see Table 3.1-6 of the ISA Summary). By assigning a number to each category of consequence (ranging from low (1) to intermediate (2) to high (3)) and likelihood (ranging from highly unlikely (1) to unlikely (2) to not unlikely (3)), a qualitative risk index can be calculated for each combination of consequence and likelihood. Unacceptable risk was defined as an index of 6 or more and required IROFS to reduce the likelihood and/or consequence to a risk index of 4 or less.

Q.18. How were consequences determined?

A.18. (WT) Consequence limits are described in terms of radiological and chemical doses (from licensed material or hazardous chemicals produced from licensed material) defined in 10 C.F.R. § 70.61(b) for high consequence events and 10 C.F.R. § 70.61(c) for intermediate consequence events. It should be emphasized that these are not acceptable exposure limits for workers or members of the public. Rather, they provide an input into the facility's design, as additional safety features must be provided if an unmitigated event can result in such a consequence level. In determining the consequence, the applicant may use an approved method to calculate an estimated dose or concentration for a given event, or simply declare the event to be a high consequence. LES declared all criticality accidents to be high consequence, therefore, to meet § 70.61(d), only preventive IROFS designed to reduce the likelihood may be used for criticality accidents. In terms of LES's risk matrix, a reduction of the likelihood to

"highly unlikely" would result in a reduction of the risk index value to 3 (1 for "highly unlikely" multiplied by 3 for "high" consequence), which is an acceptable value because it is less than 4 on the risk matrix.

Q.19. How were initiating event and IROFS failure frequencies determined?

A.19. (WT) The initiating event may be an IROFS failure or some event external to the process node being analyzed. The likelihood of failure was qualitatively evaluated for each IROFS, often based on the operational history of similar facilities. While much of that operational history is based on over 30 years of operation, the staff recognizes that history includes well over 100,000 machines and all of the associated supporting operational and maintenance activities, which are well defined.

Q.20. How did LES define highly unlikely and unlikely?

A.20. (WT) LES developed definitions for the terms "highly unlikely," "unlikely," and developed three categories according to likelihood which were applied to initiating events and IROFS failure frequencies:

- Category 1 Highly Unlikely has a probability of occurrence of less than 10^{-5} per event per year.
- Category 2 Unlikely has a probability of occurrence of between 10^{-4} and 10^{-5} per event per year.
- Category 3 Not Unlikely has a probability of occurrence of more than 10^{-4} per event per year.

Q.21. How did LES address the requirements relating to the NCS program in the application?

A.21. (HF) In Chapter 5.0 of the LES License Application (i.e., LES refers to this as the Safety Analysis Report), LES described programmatic commitments and descriptions on how it would meet those commitments related to the NCS program. The areas that LES

addressed for NCS included: Regulatory Guides and American Nuclear Society-8 standards that would be used, the program for management of the NCS program, the methodologies and technical practices that would be followed, the criticality accident alarm system, the means for ensuring subcriticality of operations including the margin of subcriticality for safety, and baseline design criteria. Previously, some licensees provided specific details about the results of having an NCS program (i.e., design of equipment, very specific controls similar to Technical Specification Requirements for nuclear power plants). However, with the addition of Subpart H to make 10 C.F.R. Part 70 even more risk-informed and performance-based, that is not the approach that NRC expects to see in a license application. Therefore, NUREG-1520 was written with the assumption that the applicant or licensee would provide in the license application the commitments and descriptions of how to meet those commitments. This is the approach used by LES.

Q.22. How will the commitments regarding the NCS program be implemented?

A.22. (HF) As with all 10 C.F.R. Part 70 facilities, the NCS program sets forth the commitments and descriptions of how to meet those commitments to ensure that facility design and operations will remain subcritical under both normal and credible abnormal conditions. LES has done this in two ways. For single parameter limits, LES established limiting values for parameters (these were in Tables 5.1-1 and 5.1-2 in the application and were the basis for SER Tables 5.3-1 and 5.3-2) using k_{eff} calculations. As appropriate, these are applied to the buildings, systems, or components of the facility. For some components, those limits are not operationally acceptable and so, LES performed specific k_{eff} calculations. In either case, the limits were developed such that the calculated k_{eff}, is lower than the k_{eff} limit in the license application with an acceptable margin of subcriticality. These controls and the rest of the commitments and descriptions in the license application will ensure that the NEF will remain subcritical under both normal and credible abnormal conditions and will have an effective NCS

program.

Q.23. You refer to the margin of subcriticality. Please explain this concept.

A.23. (HF) The term is used in 10 C.F.R. § 70.61(d) and states that an applicant must ensure that all nuclear processes remain subcritical, including use of an approved margin of subcriticality for safety. There are two ways that an applicant can demonstrate subcriticality. An applicant may (1) demonstrate that single parameter limit values are appropriate or (2) perform a specific criticality calculation for k_{eff}. Using method (1), the margin referred to is a percentage difference between what is known to be critical and what the applicant proposes to use (see percentage values in SER Table 5.3-1, Staff Exhibit 49-M). Using method (2), the margin referred to is an administrative margin that an applicant proposes to use (LES chose 5%) as part of the basis for the k_{eff} equation.

Q.24. What margin did LES propose to use?

A.24. (HF KM) Using method (1), LES calculated the percentage values in License Application Table 5.1-1 by comparing the 5 wt.% U^{235} and 6.0 wt.% U^{235} single parameter limit values from k_{eff} calculations (75% for volume, 90% for cylinder diameter, 86% for slab thickness, 72% for mass with no double batching, 45% for mass with double batching). Using method (2), LES used an administrative margin of 5%, consistent with NRC guidance documents (NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," March 1997, and NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," January 2001) that indicate, for LEU fuel cycle facilities, a 5% administrative margin, and a k_{eff} equation of k_{eff} = calculated k_{eff} + 2 times (uncertainty in the calculated k_{eff}) # 0.95, should be adequate.

Q.25. How did LES demonstrate that this was an appropriate margin for calculations?

A.25. (HF) LES followed the approach outlined in NUREG-1520, section 5.4.3.4.(8)(g), which states that an applicant should prepare a validation and verification report describing the

bias, uncertainty in the bias, uncertainty in the methodology, uncertainty in the data, uncertainty in the benchmark experiments, and margin of subcriticality for safety, as well as the basis for these items and supplemented that analysis with a qualitative argument regarding the low facility NCS risk.

Q.26. What is the purpose of the verification aspect of the report?

A.26. (HF) Verification is the process by which the same computer code input files are run on different computers, using the same computer code options, and then compared to determine whether the results are similar. The input files chosen need to be representative of the facility. For a probabilistic computer code like the one used by LES (MONK8a, Monte Carlo computer code), in which random numbers are used, the results need to be statistically equivalent for the computer code to be verified.

Q.27. What is the purpose of the validation aspect of the report?

A.27. (HF) Validation is the process (including the methodology, data, and calculations) by which the applicant performs a statistical analysis in which critical experiments similar to actual or anticipated facility conditions are chosen by the applicant and then analyzed to determine in one or more equations, the USL. The validation process needs to take into account assumptions in the methodology, administrative margin, uncertainties and biases in the data, and penalties for not having enough data to cover the area of applicability (AOA).

Q.28. What is the bias?

A.28. (HF) The bias is a measure of the systematic differences between experimental data and calculational results. The bias may be expressed as positive when the calculations produce greater values than those obtained from experiments. When the results of the calculations are lower than those from experiments, the bias is negative.

Q.29. Is the NCS Validation and Verification (V&V) Report reviewed by the Staff?

A.29. (HF) Yes. The V&V report is used by the NRC NCS reviewer when determining

whether the k_{eff} equation that the applicant proposes to commit to in the license application is acceptable. The applicant provides a summary of the V&V report (e.g., methodology, data, results) in the License Application. The Staff reviews the summary information of the V&V report to determine if it is reasonable and meets the margin of subcriticality for safety requirement for calculations in § 70.61(d). The V&V report is not part of the license application.

Staff's Review

Q.30. Mr. Troskoski, were you the primary Staff reviewer of the LES ISA Summary submitted with the LES License Application?

A.30. (WT) Yes. However, it is important to note that my review was complemented and supplemented by the Staff NCS reviewer, Harry Felsher, as well as the other Staff reviewers in other safety disciplines.

Q.31. Where is your review documented in the SER?

A.31. (WT) My review is documented in Chapter 3.0 of the SER.

Q.32. Please explain how you conducted your review of the applicant's ISA Summary.

A.32. (WT) My review of the applicant's ISA Summary consisted of two basic approaches. First, I reviewed the proposed ISA program commitments, including the ISA methodology, to assure that they met the regulatory requirements. By comparing the applicant's commitments to the regulatory requirements in 10 C.F.R. § 70, Subpart H, and utilizing the guidance provided in NUREG-1520, Chapter 3.0, "Integrated Safety Analysis (ISA) and ISA Summary," and Appendix A, "Example Procedure for Accident Sequence Evaluation," I determined that the ISA Summary met the regulatory requirements. I further determined that the program commitments were consistent with the guidance contained in NUREG-1520 and were, therefore, acceptable.

Since adequate implementation of these requirements and commitments is necessary to assure adequate safety, the second part of my review consisted of performing a vertical slice review of selected accident scenarios to confirm that the ISA Summary was adequately implemented. This part of my safety determination relied on both the regulatory guidance and my 32 years of professional experience in the nuclear field. I focused on the system description and diagrams as I followed the accident scenario descriptions, IROFS descriptions, and application of the applicant's ISA methodology.

Based on training that I have received in the ISA analysis method selected by the applicant (HAZOP), tours at the Almelo facility in The Netherlands upon which the applicant is basing its design, and my past experience in conducting the safety review of the Lead Cascade and the ongoing review of another proposed gas centrifuge uranium enrichment facility, as well as ISA reviews of three LEU fuel fabrication facilities and the proposed MOX facility, I determined that the applicant had performed an adequate ISA and documented the results in the ISA Summary.

Q.33. Please describe how you conducted a vertical slice review of an accident scenario.

A.33. (WT) I reviewed all of the chemical and many of the NCS accident sequences listed in Table 3.7-1 of the ISA Summary, which is entitled, "Accident Sequence and Risk Index." This table lists all of the accident sequences identified by the applicant's ISA Team that had unmitigated consequences exceeding the performance requirement consequence levels listed in § 70.61(b) and (c). I compared those accident sequences with the process descriptions and diagrams contained in Section 3.4 of the ISA Summary. Based on that review and my knowledge of the gas centrifuge uranium enrichment process of several different plants, I determined that there was reasonable assurance that the applicant had identified all of the hazards that could affect radiological safety and the accident sequences that could exceed

the performance requirements. I also looked at selected examples contained in Table 3.7-3 entitled, “External Events and Fire Accident Sequences and Risk Index.”

I then reviewed the IROFS assigned by the applicant in Table 3.7-1 and the indices assigned and confirmed that the assigned values would reduce the risk to an acceptable level. My review of LES’s assessment of the likelihood of failure or success of safety controls was qualitative. For this type of facility, the basis for assessing this element of risk can be supported by operating experience, industry data or expert engineering judgement. Unlike reactors which have a probabilistic risk analysis requirement that requires a quantitative evaluation, fuel facilities are permitted by the regulations to perform qualitative assessments of likelihood.

To assure that the assigned IROFS were reasonable for their intended function, I reviewed Table 3.7-2, entitled, “Accident Sequence Descriptions.” This table identifies each IROFS used in each accident sequence and the assigned indices. I reviewed the accident descriptions and confirmed that the sequence was adequately described such that the function of each specific IROFS could be understood, and that the IROFS were reasonable for that accident sequence. Furthermore, I also considered the application of management measures designed to ensure the reliability and availability of IROFS, as described in section 3.3.3.1.3 of the SER, the application of an NQA-1 program to all IROFS, and the utilization of the applicant’s “IROFS Boundary Definitions.” It should also be noted that certain IROFS required “enhanced” administrative controls or that certain automatic engineered controls have a high availability. In these cases, the bases for these additional requirements is provided in section 3.8.3 of the SAR.

Q.34. Please walk through an accident scenario to demonstrate how your review was conducted.

A.34. (WT) I will select two examples, one for chemical safety and another for

criticality safety.

Chemical Safety

For chemical safety, the largest inventory of hazardous material, UF_6, is located in a 14-ton feed cylinder. Loss of this confinement barrier could result in a significant release of hazardous material if the UF_6 is in a liquid state. Consequently, this would be a bounding accident.

From accident scenario UF1-1, described in Table 3.7-2, we note that the initiating event is a failure of the solid station heater controller that causes it to remain on. The cylinder overheats and hydraulically ruptures. For the uncontrolled accident sequence, the consequences are assumed to be high. Table 3.7-1 assigns an initiating event index of -2 (based on no failures in over 30 years), and a total likelihood index of -2, as there are no assumed preventive or mitigative measures. The likelihood index of -2, cross referenced in Table 3.1-8, yields a likelihood category of 3 ($-4 < T$). Since the assigned consequence category is 3 (high), the total risk index is determined by multiplying the likelihood and consequence indices, which yields a 9. Table 3.1-6 identifies a 9 index as unacceptable. Therefore, IROFS are required.

The applicant identifies IROFS 4 and 5 for this accident scenario. From Table 3.8-1, we see that IROFS 4 is an automatic trip of the station heaters on high cylinder temperature that is performed by a hard-wired temperature sensor for an automatic, fail-safe trip. IROFS 5 is an automatic trip of the station heaters on high station internal air temperature that is performed by a capillary temperature sensor that will be automatic, failsafe, independent and diverse from IROFS 4. Each IROFS is assigned a failure probability index of -2, which corresponds to a single active engineered control.

With application of the two IROFS, the total likelihood index becomes -6 (-2 initiating event frequency, plus -2 for each of the two preventive IROFS). The -6 corresponds to a new

likelihood category 1, or highly unlikely. Multiplying the consequence category 3 by the likelihood category 1 yields an overall risk index of 3, which is an acceptable result per table 3.1-6.

I qualitatively considered the accident sequence and results to determine the reasonableness of the outcome. In this scenario, a heater controller has an initiating event frequency of -2 (no failures in over 30 years), which is reasonable. Two independent, fail-safe active engineered controls are provided to terminate the energy source to the heaters. Additionally, a conservative setpoint would be able to provide a sufficient system response time due to the mass and heat capacity of the UF_6 being heated by a hot air source.

Criticality Safety

The largest unisolable inventory of enriched UF_6 would be in a Mark 48Y 14-ton product cylinder. For an uncontrolled accident sequence, the initiating event is a Mark 48Y cylinder of enriched UF_6 placed in a feed station, causing an enrichment higher than license limits. It is assumed that an inadvertent criticality occurs, resulting in high consequences.

From accident scenario PT2-2, described in Table 3.7-2, we note that the initiating event is a failure of IROFS 6a, whereby an operator fails to distinguish between the visual markings of cylinders in the UF_6 area to ensure that filled product cylinders are not placed on-line. For the uncontrolled accident sequence, the consequences are assumed to be high. Table 3.7-1 assigns an initiating event frequency of -1 (which corresponds to an administrative IROFS with a large margin), and a total likelihood index of -1, as there are no assumed preventive or mitigative measures. The likelihood index of -1, cross referenced in Table 3.1-8 yields a likelihood category of 3 ($-4 < T$). Since the assigned consequence category is 3 (high), the total risk index is determined by multiplying the likelihood and consequence indices, which yields a 9. Table 3.1-6 identifies a 9 index as unacceptable. Therefore, IROFS are required.

The applicant identifies IROFS 7 and 6b for this accident scenario. From Table 3.8-1,

we see that IROFS 7 is a design feature to physically prevent a product cylinder from being placed in a feed station (i.e., a passive engineered control). IROFS 6b requires the administrative verification of the ^{235}U concentration prior to placing the cylinder on-line. The failure index of IROFS 7 is a -3, representing a single passive engineered control. The failure index of IROFS 6b is -2, which corresponds to an administrative IROFS for a routine planned operation.

With application of the two IROFS, the total likelihood index becomes -6 (-1 initiating event frequency, plus -3 for IROFS 7 and -2 for IROFS 6b). The -6 corresponds to a new likelihood category 1, or highly unlikely. Multiplying the consequence category 3 by the likelihood category 1 yields an overall risk index of 3, which is an acceptable result per table 3.1-6.

I qualitatively considered this accident sequence and results to determine the reasonableness of the outcome. An initiating event frequency of -1 assumes a few failures during the lifetime of the facility. Since this process set will be carried out by trained and qualified operators in accordance with approved procedures, and the cylinders will be distinctively marked for visual identification, the -1 index is conservative. IROFS 7 will be a passive control that will physically prevent the cylinder from being loaded. Finally, IROFS 6b will be the routine assay sampling of each cylinder prior to placing the cylinder on-line. Further, there would be no financial or production reason for an operator to attempt such an evolution. Together, this strategy provides reasonable assurance that a product cylinder will not be placed on-line to the cascade.

With regard to safe-by-design, the ‘loss of a safe-by-design attribute’ is an accident sequence identified in Table 3.7-1 of the ISA Summary. The initiating event index for the ‘loss of a safe-by-design attribute’ for the components described in Tables 3.7-6 through 3.7-21 is assigned a value of -5. This -5 index corresponds to a likelihood category of 1 (highly unlikely).

Assignment of the -5 initiating event index is based on the fact that safe-by-design attributes do not rely on a human interface to perform their criticality safety function. The only potential means to cause failure of a safe-by-design attribute would be to implement a design change. In this regard, these safe-by-design attributes are passive features subject to the applicant's NQA-1 program commitments and the management change program required under 10 C.F.R. § 70.72.

The applicant provided a qualitative evaluation of potential mechanisms that could impact the criticality safety function of the safe-by-design attributes (see Tables 3.7-6 through 3.7-21), but found that these mechanisms were not credible. Based on my knowledge of the process and operating parameters, I qualitatively determined that this approach was reasonable.

Q.35. What were your findings regarding the ISA Summary?

A.35. (WT) I found that the applicant performed an ISA to identify and evaluate hazards and potential accidents, as required by the regulations. The ISA Summary and other information provide reasonable assurance that the applicant identified IROFS and established engineering and administrative controls that ensure compliance with the performance requirements. The ISA results, as documented in the ISA Summary, provide reasonable assurance that the failure of safe-by-design attributes will be highly-unlikely and that IROFS, management measures, and the applicant's programs, if properly implemented, make all credible intermediate consequence events unlikely, and all credible high consequence events highly unlikely.

Q.36. Mr. Felsher, were you the primary criticality safety reviewer for the Staff of the LES license application?

A.36. (HF) Yes.

Q.37. Where is your review documented?

A.37. (HF) In Chapter 5.0 of the SER.

Q.38. Please explain how you conducted your NCS review.

A.38. (HF) I reviewed the License Application and ISA Summary, including all revisions, and other NCS-related documents that were submitted or reviewed on-site. In addition I participated in discussions about the review with LES via the following: (a) in-office-review in Massachusetts; (b) site visit to a Urenco facility; (c) multiple meetings with the applicant; (d) multiple in-office-reviews in Washington, D.C.; and (e) multiple telephone conversations.

Q.39. What portions of the License Application did you review?

A.39. (HF) I reviewed the entire License Application for elements related to NCS. These elements included: (a) Chapter 1.0 related to the applicant's requested type, quantity, and form of special nuclear material; (b) Chapter 2.0 related to qualifications and responsibilities of NCS personnel and how NCS fits into the organization; (c) Chapter 3.0 related to NCS information regarding the general and NCS-specific ISA methodology as well as the NCS information in the ISA Summary; (d) Chapter 5.0 related to the NCS Program; (e) Chapter 8.0 related to NCS information regarding the Emergency Plan; (f) Chapter 11.0 related to NCS information regarding the management measures; and (g) Appendix A related to the NCS information regarding the Quality Assurance Program. I also reviewed the entire ISA Summary for elements related to NCS. These elements included: (a) Section 3.1 related to the general and NCS-specific ISA methodology; (b) Section 3.3 related to NCS information in the facility description; (c) Section 3.4 related to the NCS information in the process descriptions; (d) Section 3.6 related to the NCS process hazards; (e) Section 3.7 related to the NCS accident sequences (i.e., initiating event, IROFS, management measures) as well as NCS safe-by-design components; and (f) Section 3.8 related to NCS IROFS.

Q.40. Did you review the ISA methodology used by LES?

A.40. (HF) Yes, the ISA Coordinator (W. Troskoski), and all the reviewers, including myself, reviewed the ISA methodology used by LES. This included a review of the index value scheme, including the definitions of the index values for IROFS and initiating events. The Staff determined that the index value scheme for IROFS and initiating events were reasonable and could be used by LES when performing the ISA and ISA Summary. This is because the LES ISA methodology was consistent with the ISA methodology described in Appendix A, "Example Procedure for Accident Sequence Evaluation," of NUREG-1520.

Q.41. What was the nature of your review of the ISA Summary?

A.41. (HF) My review was focused on Sections 3.6 (Process Hazards), 3.7 (Accident Sequences), and 3.8 (IROFS). The other parts of the ISA Summary were reviewed in order to understand the processes relevant to NCS and to ensure consistency with Sections 3.6, 3.7, 3.8, and the License Application. Similar to Mr. Troskoski's review, in Sections 3.6, 3.7, and 3.8, I reviewed: (1) the description of the accident sequences for reasonableness of clarity, accuracy, and completeness; (2) the reasonableness of appropriate IROFS for the associated accident sequence; (3) the IROFS for reasonableness of clarity and accuracy; (4) the index values of the IROFS for reasonableness; (5) the index values of the initiating event for reasonableness, and (6) the reasonableness of the management measures associated with the IROFS. In my evaluation, I took into account the accident sequence, initiating event(s), IROFS, and management measures together and determined that the ISA methodology was used appropriately and that, taken as a whole, the description of the accident sequences (i.e., initiating event(s), IROFS, index values, management measures) were reasonable. In addition, I reviewed the ISA methodology for determining that failure of safe-by-design components was highly unlikely as well as the original classified information that was submitted by LES to demonstrate that the safe-by-design ISA methodology was followed. I concluded that the safe-by-design methodology was reasonable and that the information in the original classified

information related to NCS calculations demonstrated that LES followed the methodology.

Q.42. What was the nature of your review of the other information relating to LES' ISA?

A.42. (HF) Besides the License Application and the ISA Summary, I reviewed additional information that supported the LES ISA. I reviewed three generic NCS analyses and a document with single parameter limit calculations. I reviewed a sample of the hazard analyses that described all the accident sequences. I reviewed the information in the documents qualitatively to determine if they were reasonable. I reviewed the calculations to determine if they appeared reasonable. I reviewed the classified information submittal to determine whether the criticality calculations for the safe-by-design components met the definition of safe-by-design and thus, the failure of the components were highly unlikely and § 70.61(b) was met.

Q.43. Did you review any k_{eff} calculations?

A.43. (HF) Yes, I reviewed k_{eff} calculations in documents that supported the ISA. This review included the original calculations supporting the classification of components as safe-by-design. I reviewed the underlying assumptions, calculational methods, and results and determined that, using expert judgment as a qualified NRC NCS License Reviewer, the calculations were reasonable. In this manner, I determined that LES was properly implementing the methodology for calculating k_{eff} and setting appropriate limits to ensure that operations are subcritical under normal and abnormal conditions. LES documents all k_{eff} calculations and keeps them on-site where they will be available for review by the NRC.

Q.44. How did you determine that the results of calculations are reasonable?

A.44. (HF) For the calculations concerning single parameter limits (e.g., Table 5.1-1 and 5.1-2 of the License Application), I compared the values in the tables with the values in the tables of ANSI/ANS-8.1-1996, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors." For some of the values, I interpolated the data. This is consistent with the

information on page 5-14 in the SER dated June 2005 which stated, "NRC determined that the applicant's values in Table 5.3-1 [same table as Table 5.1-1 of the License Application] are consistent with the values in ANSI/ANS-8.1 (ANSI/ANS, 1998a)." For the other calculations that I reviewed, I looked at the assumptions, calculational methods, and results. Based on my expert judgment as a qualified NRC NCS License Reviewer, I determined that the assumptions, calculational methods, and results were reasonable.

Q.45. Did you review other records that were submitted to NRC?

A.45. (HF) Yes, I reviewed three versions of the Validation and Verification (V&V) report submitted by LES. From my review of the V&V report submitted December 20, 2005, LES Exhibit 126M, which is the subject of Board questions below, I identified issues that were addressed by LES in the revision of the V&V report submitted on February 16, 2006, LES Exhibit 127M. One of the issues that I identified in the earlier report was the inclusion of reference (benchmark) experiments involving high-enriched uranium (HEU), when those experiments are not directly applicable to the operations at the NEF, which involve only low-enriched uranium (LEU). This issue has been satisfactorily addressed by LES in the most recent report by eliminating the HEU experimental data and including additional LEU benchmark experiments.

Q.46. Did you review the validation report for the purpose of determining whether LES had appropriately accounted for bias?

A.46. (HF) Yes. In the License Application, LES stated that it had validated the computer code considering 36 LEU solution experiments and found an overall positive bias (meaning that the outputs of criticality calculations were higher than the experimental results). LES did not take credit for the positive bias and conservatively assumed that it was zero. LES included both low- and high-enrichment experiments, so, the validation report (and the bias determination) applied to a broad range of hydrogen-to-uranium ratios (from 0.103 to 1378). I

found the LES approach of setting the bias to be zero for all processes and components at the NEF to be acceptable because it is consistent with NRC guidance in NUREG/CR-6698, LES Exhibit 131-M.

Q.47. How did the visit to the Urenco facility in the Netherlands inform your NCS review?

A.47. (HF) During the visit, I toured the facility, participated in discussions with Urenco and LES staff, and reviewed Urenco records related to NCS. The tour was extremely helpful because it demonstrated how simple the operation of the facility was and how few people were needed to operate the facility safely, and it provided insight into Urenco's approach to NCS. Urenco staff presented information regarding equipment operating experience and failures. LES staff presented its proposed approach to NCS for the NEF. I reviewed the classified information regarding NCS for certain operations. My review was to determine whether the information available at that time in that location was reasonable and whether it supported the ISA Summary.

Q.48. What were your findings regarding the LES NCS program?

A.48. (HF) My findings regarding the LES NCS program are on page 5-37 of the SER dated June 2005, which stated, "Based on this NCS review, the staff concludes that the applicant's NCS program meets the requirements of [10 C.F.R.] Part 70 and provides reasonable assurance for the protection of public health and safety, including workers and the environment."

Response to Board Questions

Q.49. <u>Question 5 from the Board's January 30, 2006 Order:</u>

> From Table 7-3 of the Monk 8 Verification/Validation report, revision 1, the Board sees that the criticality calculations for the items relied on for safety (IROFS) concerning pipe works involve hydrogen to uranium (H/U) ratios from 12 to 14. How does the

> staff compute the bias allowance for these cases, given the spreads indicated in Figure 6.3 of that report? Is the number in the Safety Evaluation Report (SER) correct?

A.49. (WT, HF, KM) LES, which was responsible for the preparation of the validation report, will address the bias issues raised by the Board in its pre-filed testimony.

Q.50. Question 6 from the Board's January 30, 2006 Order:

> How does the staff justify acceptance of IROFS for depleted uranium hexafluoride (UF_6) mixtures with no hydrogen (except in the reflector) when, according to the second full paragraph in section 6.1 (page 29) of the report, the H/U ratio varied between 0.102 to 1378 in the calculations used for verification?

A.50. (WT, HF, KM) The variation in the H/U ratios referenced in the Board's question is related to an issue brought to LES's attention by the staff (see Answer 45). Accordingly, LES has addressed this issue and has provided an explanation of H/U variation in its pre-filed testimony.

IROFS are required for all unmitigated accident sequences identified by the applicant as exceeding performance requirements. These accident sequences are listed in ISA Summary Table 3.7-1 entitled, "Accident Sequence and Risk Index," and are described in Table 3.7-2 entitled, "Accident Sequence Descriptions." No criticality accident sequence involving depleted uranium was identified by the applicant in these tables. Consequently, the applicant developed no nuclear criticality safety (NCS)-related IROFS for any depleted uranium process. The staff concurs with the applicant's evaluation because there is no credible process in the proposed facility that could bring a depleted uranium system to a critical state (e.g., no graphite or heavy water moderated configurations). Further, while there are IROFS that address the chemical safety concerns associated with UF_6, these IROFS are independent of the degree of uranium enrichment. These IROFS protect against the chemical hazards associated with UF_6 and its chemical reaction products, including HF.

Q.51. Question 7 from the Board's January 30, 2006 Order:

> The Staff is requested to correlate the IROFS discussed in the SER with the cases listed in Table 7-3 of the report. Are all IROFS adequately represented in the table?

A.51. As discussed above, the purpose of the verification portion of the V&V Report, in which Table 7-3 is included, is to ensure that the results of running the computer code on two different machines are statistically equivalent. The Staff's review of the verification portion of the V&V Report focused on the paired k_{eff} results listed in Table 7-3 and whether those paired results were statistically equivalent. For the purposes of verification, the significance of the input files used to generate the k_{eff} results in Table 7-3 is that they are identical for each pair of results and generally represent the facility. As is the case for any verification review, the Staff's review was limited to the verification process.

Table 7-3 does not include IROFS or provide an indication of IROFS. The Staff's review of IROFS occurred during the review of the ISA Summary and addressed whether the accident sequences (i.e., initiating event, IROFS, and management measures) were reasonable. The Staff's NCS review was focused on the NCS program that will ensure the NEF will be subcritical under normal and credible conditions.

NRC recognizes that the input files chosen by LES in Table 7-3 of the V&V report represent NCS scenarios. However, there are many possible IROFS for an NCS scenario. Therefore, it is not possible to determine a specific IROFS from Table 7-3.

The Staff reviewed the values in Tables 5.1-1 and 5.1-2 of the License Application against values in standards endorsed by NRC and the Staff considered these values to be appropriate.

Q.52. Does this conclude your testimony?

A.52. Yes.

Resume for Mr. William Troskoski

QUALIFICATION PROFILE

EXPERIENCE/SKILLS

Mr. Troskoski has 30-years of nuclear experience ranging from reactor operations through the fuel cycle front end. He was a shift supervisor for a DOE heavy water production reactor, an NRC inspector qualified on both the BWR and PWR series reactors, and a Senior Resident Inspector at a duel unit PWR site. His experience includes pre-operational, startup testing and plant operations. He served as a Regional Coordinator in the Deputy EDO's Office and a Senior Enforcement Specialist in the Office of Enforcement. During the last eleven years, Mr. Troskoski has been involved in all phases of fuel cycle inspection and licensing process.

EDUCATION

Bachelor of Science Degree in Chemical Engineering under the Cooperative Program, University of Maryland, 1973.

ACCOMPLISHMENTS/STRENGTHS

Certified Reactor Shift Supervisor at Savannah River Plant 1974-1980.

Senior Resident Inspector 1981-1987.

Meritorious Service Award 1998.

PROFESSIONAL EXPERIENCE

2002 to present — Senior Chemical Safety Technical Reviewer

Responsible for the conduct of license application acceptance reviews and in-depth license application safety reviews in the areas of chemical safety, management measures, quality assurance and integrated safety analysis for the Mixed Oxide Fuel Fabrication Facility, the USEC Lead Cascade, the LES National Enrichment Facility, and the USEC American Centrifuge Plant.

Provided chemical engineering technical assistance to the Office of Investigations and other Federal agencies for a potential wrong doing case involving Hunt valves used on UF_6 cylinders.

Developed and taught several NRC internal fuel cycle training courses.

1993 to 2002	Senior Chemical Safety Fuel Cycle Inspector
	Responsible for the development of the Chemical Safety Inspection Program for NRC licensed fuel cycle facilities, including low enriched uranium fuel fabricators, high enriched uranium fuel fabricators, the USEC Gaseous Diffusion Plants (enrichment), and uranium conversion.
	Served as the lead chemical safety inspector responsible for scheduling and implementation of the routine inspection program in coordination with the Regional Offices.
	Developed Operational Readiness Review Inspection plans and served as the team leader for the restart of the Nuclear Fuel Services high enriched fuel facility and the initial certification of the USEC Gaseous Diffusion Plants at Portsmouth, Ohio and Paducah, Kentucky.
1988 to 1993	Senior Enforcement Specialist
	Responsible for the processing and coordination of reactor and fuel cycle escalated enforcement actions, including Proposed Civil Penalties, Imposition of Civil Penalties, and other related Orders. Coordinated actions with the Regional Offices, Program Office, OGC, and OI, when applicable.
1987 to 1988	Regional Coordinator - Deputy EDO's Office
	Monitored issues and emerging safety problems for licensees in Region II. Briefed the Deputy EDO as necessary.
1981 to 1987	Senior Resident Inspector
	Conducted safety inspections at a duel unit PWR. One unit conducted an extended outage to perform TMI-related modifications and return to power operations. The second unit completed construction, pre-operational testing and initiated startup testing prior to commercial operations. Supervised other resident inspectors.
1980 to 1981	Reactor Inspector - Region I
	Performed pre-operational and startup testing inspections at both BWRs and PWRs.
1974-1980	Reactor Shift Supervisor - Savannah River Plant
	Supervised reactor operations for a heavy water moderated production reactor.

Harry D. Felsher
18603 Village Fountain Drive, Germantown, MD 20874 / (301) 353-1440

RELEVANT PROFESSIONAL EXPERIENCE

U.S. NUCLEAR REGULATORY COMMISSION (NRC) ROCKVILLE, MD

Nuclear Process Engineer (Criticality) May 2000 - present
Nuclear Process Engineer (Criticality)/Project Manager March 1997 - May 2000

- Performed ~60 licensing reviews (i.e., new applications, renewals, amendments) for 10 CFR Parts 70 and 76 licensees, including inputs to requests for additional information, safety/compliance evaluation reports, and license conditions as well as participating in site visits, meetings, in-office-reviews, and teleconference calls.

- Wrote the Nuclear Criticality Safety (NCS) chapter of the fuel cycle facility standard review plan (NUREG-1520). Presented information on NUREG-1520 and the 2000 revision to 10 CFR Part 70 at American Nuclear Society (ANS) and U.S. Department of Energy NCS meetings. Wrote Revision 1 to Regulatory Guide 3.71 (NCS standards for fuels and material facilities). Wrote IN 99-20, IN 99-18, GL 98-03, and IN 97-56.

- Coordinated ~50 interdisciplinary technical and administrative licensing reviews (i.e., renewals, amendments) for 10 CFR Parts 70 and 76 licensees, including writing requests for additional information, safety evaluation reports, and license conditions as well as setting up and participating in site visits, meetings, and teleconference calls.

- Participated and led inspections as well as coordinated and wrote input to inspection reports for 10 CFR Part 70 and 76 licensees.

- Interacted with industry and members of the public concerning NRC actions related to 10 CFR Parts 70 and 76 licensing. Also, as a member of the NRC Year 2000 Task Force, interacted with international, federal, and state stakeholders.

- Qualified as both an NRC NCS License Reviewer (2001) and NRC NCS Inspector (2003) for 10 CFR Parts 70 and 76 licensees. Also, Certified as an NRC Contract Project Manager (2001) and was the Technical Project Manager for the Division's NCS contract with Oak Ridge National Laboratory for many years.

- NRC representative to consensus standards developing organizations:
  Member of ANS-8 Subcommittee and Chair of ANS-8.10 Working Group
  Former member of ANS-8.7, 8.10, and 8.19 Working Groups
  Technical Expert to U.S. Nuclear Technical Advisory Group for ISO/TC85/SC5/WG8

- Completed rotations as an NRC Division, Office, and OEDO Technical Assistant.

- Knowledge and experience in the use of:
  NCS computer codes (KENO, KENO3D, MCNP, MONK, and SCALE);
  programming languages (FORTRAN and LISP);
  computers (Mainframe, Personal, VAX, and Workstations);
  operating systems (DOS, MAC, UNIX, VAX, and Windows); and

software (Corel Office and MS Office products).

- Received extensive training in regulatory, technical, and NRC-specific areas.

- Received "Outstanding" rating in annual performance appraisals for FY 2004, 2003, 2002, 2001, and 1999; and received "Excellent" rating in FY 2005, 2000, and 1998.

RESEARCH ASSISTANT UNIVERSITIES (see below)

- As an undergraduate and graduate student, assisted research reactor directors and professors in performing experiments (e.g., radiation detectors, materials irradiation) and performing research (e.g., NCS transport cart, research reactor operator advisory system, space dosimetry, sub-critical neutron detector)

- Wrote report on reactor advisory system, M.S. Report on burnup credit for transport casks, NCS analysis for UF6 cylinders, M.S. Thesis on a portable radiation shield for the space station, and NASA report on the portable shield.

AWARDS/HONORS

- Boy Scouts of America Eagle Scout, Brotherhood Member of Order of the Arrow, and Life Member of the National Eagle Scout Association, since 1983.

- NRC's sole choice for William A. Jump Memorial Foundation Award, 2003.

- Received NRC Instant Cash Awards (2001-multiple, 1998-multiple), Performance Awards (2005, 2003), Special Achievement Awards/Certificate (2000, 1998), Special Act Awards (2005, 2001), and Time-Off Award (2003).

- Received U.S. Government Year 2000 Medal/Recognition Letter/Plaque, 2000.

- Received Outstanding Service and Leadership Awards from ANS Local Section and Student Branches (2002, 1993, 1989).

- Received Best NCS Paper awards at Student ANS and ANS National Meetings, 1992.

- Member of high school team to design and build a NASA Space Shuttle "Getaway Special" experiment, 1983.

EDUCATION

THE UNIVERSITY OF TEXAS (UT) AUSTIN, TX
Studied Nuclear Engineering July 1994 - December 1996

THE OHIO STATE UNIVERSITY (TOSU) COLUMBUS, OH
M.S., Nuclear Engineering June 1994

TEXAS A&M UNIVERSITY (TAMU) COLLEGE STATION, TX
M.S., Nuclear Engineering December 1991

Harry D. Felsher
18603 Village Fountain Drive, Germantown, MD 20874 / (301) 353-1440

UNIVERSITY OF MARYLAND (UMCP) COLLEGE PARK, MD
B.S., Engineering (major in Nuclear Engineering) May 1989

LEADERSHIP/VOLUNTEER/PUBLICATION EXPERIENCE

- Member, Jewish Federation Next Generation Affinity Network Council, since 2005.
- Member, Jewish Mosaic-MD Outdoor Club Board (President, Special Event Pre-Tour Chair, Secretary), since 2004.
- Member, ANS/NCS Division Program Committee, since 2002.
- Applied for U.S. Government and NRC Leadership Programs, since 2000.
- NRC recruiter at student and national ANS meetings, since 2000.
- Wrote abstracts, organized panels, organized sessions, presented papers, and presented posters at professional meetings, since 2000.
- Member, ANS Washington, DC Local Section Executive Committee (Vice-Chair/Chair Elect, Membership Director, Secretary), since 1999.
- ANS and NRC judge at science fairs, since 1998.
- Acted as Section Chief and Team Leader, many times since 1998.
- President and other positions, ANS TOSU and UMCP Student Branches, 1984 - 1994.

PROFESSIONAL MEMBERSHIPS

- Member of Order of the Engineer, since 1991.
- American Nuclear Society
  - Member, NCS Division since 1992
  - Member, TOSU Student Branch, 1991 - 1994
  - Member, National, since 1986
  - Member, UMCP Student Branch, 1984 - 1989

KEVIN J. MORRISSEY
6122 BROOKHAVEN DRIVE
FREDERICK, MD 21701
WORK PHONE: (301) 415-6282
EMAIL: KJM@NRC.GOV

SUMMARY

As a nuclear engineer/physicist, has over 30 years of experience in the nuclear engineering analysis field. Areas of expertise include a wide variety of nuclear analysis methods, nuclear reactor operational support and licensing, reactor core design, criticality and dose rate calculations, training and supervision.

EXPERIENCE

Nuclear Process Engineer 6/04-Present
United States Nuclear Regulatory Commission
Responsible for the review of fuel cycle facility license applications and amendments, ISA Summary reviews and all aspects related to nuclear criticality safety.

Senior Technical Specialist 5/02-11/03
Framatome ANP (purchased DE&S)
Served as criticality expert for the Independent Safety Analysis (ISA) of the Louisiana Energy Services (LES) uranium enrichment plant to support a facility licensing application and ISA Summary submittal. Familiar with 10 CFR Part 70 requirements for special nuclear material as it applies to 10CFR70.62 safety programs and analysis.
Developed and applied particle transport methodologies for various applications relating to dry fuel storage and shipping designs

Senior Technical Specialist 12/97-5/02
Duke Engineering & Services (DE&S) (purchased YAEC)
Performed component activation analyses for the Fermi-1 LMFBR and NASA Plum Brook research reactor in support of decommissioning activities, shipping and disposal. Performed benchmarking of various available activation analysis methods using measured data from the Japanese Power Demonstration Reactor.
Developed a new methodology for determining analytical fixed platinum detector response for the Seabrook Nuclear Power Station power distribution surveillance requirements.

Senior Nuclear Engineer, 9/88-11/97
Reactor Physics Group, Nuclear Engineering Department
Yankee Atomic Electric Company (YAEC)
Performed activation analyses for the YNPS, Connecticut Yankee and Maine Yankee nuclear power stations in support of decommissioning activities, shipping and disposal. Provided the licensing justification and analysis for the source and dose rate characterization for the shipping of the YNPS reactor vessel and associated components, including a measurement test plan to support the analysis conclusions.
Provided technical methodology and standards review for numerous criticality calculations for spent

fuel and new fuel storage for the Maine Yankee and Seabrook nuclear power stations including fuel re-racking, fuel zoning and Boraflex evaluations. Provided technical review of licensing submittals for various fuel transport canisters and shipping casks, including both vertical and horizontal dry fuel storage configurations.

Provided analysis for and licensed a combination fixed and movable incore detection system to meet Technical Specification requirements for operability and power distribution surveillance. Supervised the development of the reactor physics core model in the YNPS core simulator, and validated the model and acceptance testing data.

Senior Engineer, 9/85-9/88

Reactor Physics Group, Nuclear Engineering Department

Yankee Atomic Electric Company

Provided project supervision and technical support for reload licensing analysis, core follow and operational support for the operation of the YNPS. Authored an YNPS-specific reactor physics-training manual for plant operators. Provided analysis and measurement test program for the benchmarking of fixed detectors installed in movable detector paths. Developed fuel management design options for extended fuel cycle operation of the YNPS lowering fuel costs.

Served as the Nuclear Engineering Coordinator for the YNPS, responsible for coordinating all reload-related work performed by the Nuclear Engineering Department, including scheduling, prioritizing and budget determination and tracking. Instituted a Core Operating Limits Report for the YNPS that expedited the licensing process for cycle dependent operation. Authored a Technical Specification change to implement the use of combination of uncertainties in determining measured linear heat generation rates (LHGRs) to improve operating margins and allow full power operation.

Engineer, 9/75-9/85

Reactor Physics Group, Nuclear Engineering Department

Yankee Atomic Electric Company

Provided project supervision and technical support for reload licensing analysis, core follow and operational support for the YNPS. Performed analysis for fuel reconstitution options prior to YNPS Cycle 15 start-up after fuel damage was detected that allowed operation within the licensed design. Performed fuel management studies to change fuel assembly component structures from stainless steel to zircaloy to save on fuel enrichment costs. Provided reactor physics training to shift technical advisors (STAs) for initial qualification

EDUCATION

BS, Mathematics, University of Massachusetts, Amherst, Mass., 1976

Graduate Courses, Nuclear Reactor Physics, Massachusetts Institute of Technology (MIT) and University of Lowell, 1979-1980

Undergraduate Courses, Introduction to C Programming, Advanced C Programming, and Networking and Communications, Worcester State College, 1999-2000.

TRAINING

Management Training Program, Bentley College

Deterministic Methods in Radiation Transport, Oak Ridge National Laboratory (ORNL)

Theory and Application of Neutron Transport Methods, University of Massachusetts Lowell

Introduction to MCNP

Modern Nodal Methods for Analyzing Light Water Reactors (LWRs), MIT

Incore Fuel Management (ICFM) Package Training, Studsvik of America

Theory of Operation of the Yankee Rowe Fixed Incore Detector System, Babcock & Wilcox
Combustion Engineering (CE) Simulator Training for Operator Qualification
Nuclear Power Reactor Safety Seminar, MIT
PWR Information Course, Westinghouse Electric Company
Quality Service Everytime, Yankee Atomic Electric Company

AWARDS/HONORS

American Nuclear Society Best Paper Award for "Determining Yankee Nuclear Power Station Neutron Activation," co-authored with K. J. Heider and personally presented at the 1993 ANS Winter Meeting.

Technical Session Chairman for Activation Analysis Methods, Radiation Protection and Shielding Topical Meeting, April 1996.

Recognized in NRC approval of implementation of fixed detectors for the Yankee Nuclear Power Station for providing excellent technical justification and presentation.

Louisiana Energy Services, L.P., Docket No. 70-3103-ML
March 2006 Mandatory Hearing on Uncontested Issues
Prefiled Hearing Exhibits

Party Exh. #	Witness/ Panel	Description
Staff 49-M	Standard Review Plan	NUREG-1827, "Safety Evaluation Report for the Proposed National Enrichment Facility in Lea County, New Mexico," (2005)
Staff 50-M	Standard Review Plan	"Louisiana Energy Services National Enrichment Facility Safety Evaluation Report Executive Summary," (Sept. 16, 2005).
Staff 51-M	Standard Review Plan	NUREG-1520, "Standard Review Plan for Review of License Applications for Fuel Cycle Facilities," (2002).
Staff 52-M	Decommissioning Funding	SECY-03-0161, "2003 Annual Update - Status of Decommissioning Program," (Sept. 15, 2003).
Staff 53-M	Decommissioning Funding	NUREG-0586, "Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," (1981).
Staff 54-M	Decommissioning Funding	NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," (1988).
Staff 55-M	Decommissioning Funding	NUREG-0584, "Assuring the Availability of Funds for Decommissioning Nuclear Facilities," (1982).
Staff 56-M	Decommissioning Funding	NUREG-CR-1481, "Financing Strategies for Nuclear Power Plant Decommissioning," (1980).
Staff 57-M	Decommissioning Funding	57 Fed. Reg. 30,383-30,387 (July 9, 1992)
Staff 58-M	Criticality	"National Enrichment Facility Integrated Safety Analysis Summary," (2004).

Party Exh. #	Witness/ Panel	Description
Staff 59-M	Criticality	Interim Staff Guidance (ISG)-03, "Nuclear Criticality Safety Performance Requirements and Double Contingency Principle," (Feb. 17, 2005).
Staff 60-M	FEIS Purpose and Need	NUREG-1790, "Final Environmental Impact Statement for the Proposed National Enrichment Facility in Lea County, New Mexico," (2005).
Staff 61-M	FEIS Purpose and Need	Louisiana Energy Services Environmental Report, Section 1.0, "Purpose and Need for the Proposed Action," (2004).
Staff 62-M	FEIS Purpose and Need	Council on Environmental Quality Regulations, 40 CFR 1500.1 and 1502.13.
Staff 63-M	FEIS Purpose and Need	Natural Resources Conservation Service, U.S. Dept. of Agriculture, "Writing a Purpose and Need Statement," (2003).
Staff 64-M	FEIS Purpose and Need	Letter from J.L. Connaughton, Executive Director, Council on Environmental Quality, to N.Y. Mineta, Secretary, U.S. Dept. of Transportation (May 12, 2003).
Staff 65-M	FEIS Purpose and Need	Maeda, H. 2005. "The Global Nuclear Fuel Market – Supply and Demand 2005-2030: WNA Market Report", World Nuclear Association Annual Symposium
Staff 66-M	FEIS Purpose and Need	Combs, J. 2004. "Fueling the Future: A New Paradigm Assuring Uranium Supplies in an Abnormal Market", World Nuclear Association Annual Symposium
Staff 67-M	FEIS Purpose and Need	Cornell, J. 2005. Secondary Supplies: Future Friend or Foe?, World Nuclear Association Annual Symposium
Staff 68-M	FEIS Purpose and Need	Van Namen, R. (2005) "Uranium Enrichment: Contributing to the Growth of Nuclear Energy", USEC Presentation to Platts Nuclear Fuel Strategies Conference.

Party Exh. #	Witness/ Panel	Description
Staff 69-M	FEIS Purpose and Need	Euratom (2005) "Analysis of the Nuclear Fuel Availability at EU Level from a Security of Supply Perspective", Euratom Supply Agency – Advisory Committee Task Force on Security of Supply.
Staff 70-M	FEIS Purpose and Need	International Energy Outlook (2000-2005)
Staff 71-M	FEIS Purpose and Need	EIA, "Uranium Marketing Annual Report," (2004), available at http://www.eia.doe.gov/cneaf/nuclear/page/forecast/projection.html.
Staff 72-M	FEIS Purpose and Need	Letter from W.D. Magwood, U.S. Dept. of Energy, to M. Virgilio, U.S. Nuclear Regulatory Commission, "Uranium Enrichment," (July 25, 2002).
Staff 73-M	FEIS Purpose and Need	U.S. Dept. of Energy, "The Global Nuclear Energy Partnership," (2006), available at http://www.gnep.energy.gov/default.html.
Staff 74-M	FEIS Purpose and Need	U.S. Dept. of Energy, "GNEP Element: Expand Domestic Use of Nuclear Power," (2006), available at http://www.gnep.energy.gov/pdfs/06-GA50035c_2-col.pdf.
Staff 75-M	FEIS Purpose and Need	U.S. Dept. of Energy, "GNEP Element: Establish Reliable Fuel Services," (2006), available at http://www.gnep.energy.gov/pdfs/06-GA50035g_2-col.pdf.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LOUISIANA ENERGY SERVICES, L.P.	)	Docket No. 70-3103
	)	
(National Enrichment Facility)	)	ASLBP No. 04-826-01-ML
	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF PRE-FILED MANDATORY HEARING TESTIMONY CONCERNING CRITICALITY" in the above-captioned proceedings have been served on the following by deposit in the United States mail; through deposit in the Nuclear Regulatory Commission's internal system as indicated by an asterisk (*), and by electronic mail as indicated by a double asterisk (**) on this 24th day of February, 2006.

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

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

Office of the Secretary * **
ATTN: Rulemakings and Adjudication Staff
U.S. Nuclear Regulatory Commission
Mail Stop: O-16C1
Washington, D.C. 20555
E-mail: HEARINGDOCKET@nrc.gov

Administrative Judge * **
Charles Kelber
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Mail Stop: T-3F23
Washington, D.C. 20555
E-Mail: cnkelber@aol.com

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

Mr. Rod Krich, Vice President
Licensing, Safety and Nuclear Engineering
Louisiana Energy Services
2600 Virginia Avenue NW.
Suite 610
Washington, D.C. 20037

James R. Curtiss, Esq. **
Dave Repka, Esq. **
Martin O'Neill, Esq. **
Amy C. Roma, Esq. **
Tyson R. Smith, Esq. **
Winston & Strawn
1700 K Street, N.W.
Washington, D.C. 20006
E-mail: jcurtiss@winston.com
drepka@winston.com
moneill@winston.com
aroma@winston.com
trsmith@winston.com

/RA/

Lisa B. Clark
Counsel for NRC Staff