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Your ref:
Our ref: LTR-RAC-17-41

September 25, 2017

SUBJECT: WESTINGHOUSE RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION
(Cost Activity Code: L33317)

Westinghouse Electric Company LLC (Westinghouse) is pleased to provide the enclosed partial responses to your Requests for Additional Information dated June 23, 2016 and November 23, 2016 regarding our License renewal application.

If you have any questions, please contact me at (803) 647-3338.

Nancy Blair Parr

Nancy Blair Parr, Manager
Licensing
Westinghouse Columbia Fuel Fabrication Facility
Docket 70-1151 License SNM-1107

Enclosure 1: Westinghouse June 2016 RAI Responses – 10 pages
Enclosure 2: Westinghouse November 2016 RAI Responses – 21 pages

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Enclosure 1

Westinghouse June 2016 RAI Responses

REQUEST FOR ADDITIONAL INFORMATION	REG BASIS	WESTINGHOUSE RESPONSE
RAI 41 Discuss how procedures of activities involving licensed material and IROFS are prepared, authorized, and approved for distribution, when multiple disciplines are involved. Section 3.4.1 states that activities involving licensed material and IROFS are conducted in accordance with properly issued and approved procedures.	10 CFR 20.1101(b)	No change made to the License Application. When a modification is designed per the CFFF Configuration Management Program, the "change" is reviewed by the various safety disciplines to assure that 10CFR70.72 requirements are met. When a "change" potentially has an impact on existing IROFS or requires new IROFS, the procedures for those IROFS must be approved and issued prior to implementation and start-up of the "change." The CFFF Configuration Management Program assures that the CFFF maintains control of changes to procedures and requires multi-disciplinary review and approval to ensure procedures continue to meet their specification requirements and comply with all applicable regulations. Procedures are prepared, authorized, and approved in accordance with established guidelines for the Electronic Training and Procedure System (ETAPS), as noted in CFFF procedure CA-002.
RAI 42 Provide the minimum training requirements for the Radiation Safety Function Manager. Provide the training requirements for Regulatory Function Engineers working in the Radiation Safety Function. Section 2.1.1.3 describes minimum requirements for a Safety or Regulatory Component Manager as well as Regulatory Function Engineers.	10 CFR 70(a)(6)	No change made to the License Application. Section 2 describes the Management Organization, including specification of structure, responsibilities and authorities. Section 2.1.1.3 (e) covers the details specific to Regulatory or Safety Component Managers and Engineering Qualifications. Additional information is detailed in Section 3.4.2.2. At the CFFF, there is a training and qualification program for EH&S personnel. Training records are maintained and are available onsite for NRC review and inspection.
RAI 43 Identify refresher training or requalification requirement for engineers, operating technicians, and process operators, above the annual refresher training requirement for radiation workers. Section 3.4.2.2 through Section 3.3.2.4	10 CFR 19.12	No change made to the License Application. Refresher training varies between departments and functions Refresher and requalification training is determined using a systematic approach. Examples are: Periodic refresher training for personnel performing administrative IROFS is performed. This periodic review and refresher training assures the availability and reliability of

describe additional training and qualification requirements for engineers, operating technicians, and process operators.		administrative IROFS and is considered part of the "procedure" management measure described in Section 3.4 of the License Application. Also, any time changes are made to processes/procedures, training is required for those processes/procedures. Periodic requalification for operators performing activities relied on for safety requires that an OJT Trainer determines if a re-qualifying person can continue to successfully and safely perform the process per the procedure(s). In the license application, the use of Electronic Training Check Lists (ECL) is required for operator training. Where Process Operation Qualification Interlocks are in use, then an operator cannot perform a task until he/she is current on procedure, training and qualification requirements. Nuclear Safety Qualification Training (NSQT) is an example of this. Procedures, training and qualification requirements assure that activities related to IROFS and management measures are properly performed in accordance with CFFF specifications. In addition, if disqualified, workers will be re-qualified for the procedure/process by training on the processes/procedures again.
RAI 44 Describe survey requirements, procedures or surveys performed, for personnel or areas to detect contamination and any limits associated. Section 5.2.29 describes the CFFF contamination control program. Contamination survey limits and survey frequencies are shown in Table 5.1. Survey limits are provided only for smear (alpha contamination) surveys.	10 CFR 20.1501(a)(2)(iii)	No change made to the License Application. Surveys are conducted throughout the facility as required per the license and flow down procedures specific to the plant operations. Limits and survey frequency vary depending on the potential for contamination. See ROP-05-014 and associated forms (ROF-05-014-xx) for survey procedure, limits, and locations.
RAI 45 Discuss how the radiation protection training program is included in triennial assessment of the radiation safety program. Describe how the effectiveness of training and instructors is evaluated. Section 5.2.67 states the entire is assessed on a triennial frequency.	10 CFR 20.2102(a)(2)	No change made to the License Application. A triennial audit of the Radiation Program described in Chapter 5.0 of the License Application includes assessment of training for ALARA principles and requirements as stated in Section 5.2.3, sub bullet 6. In addition, the CFFF training program is described in Section 3.4 of the License Application and is also assessed in the triennial management measures audit of Chapter 3.0.
RAI 47 Describe the effectiveness of ventilation systems as they relate to public	10 CFR 20.1101(d)	No change made to the License Application.

exposure. Section 5.2.3 describes several actions taken by the CFFF to maintain exposure to the public ALARA.		Ventilation systems exhausted to the atmosphere are continuously sampled for radioactivity. If action levels are exceeded, actions are taken to mitigate releases to the environment and subsequent dose to the public. There is an administrative limit for dose to the public due to effluents that is reviewed and approved by the ALARA committee annually. This year it is 1 mrem. See RA-219-1.
RAI 52 Discuss the responsibilities of the chemical safety component within NRC's regulatory jurisdiction. Chapter 2, Organization, presents an organization chart (Figure 2.2) that identifies the safety components. Section 2.1.1.3(b) lacks a discussion of responsibilities of the chemical safety component within NRC's regulatory jurisdiction. 52.1. Describe the activities or responsibilities included under the Chemical Safety Component. Discuss compliance with the NRC chemical safety requirements (e.g., chemical safety aspect of the ISA, chemical safety IROFS, procedures, training, audits, as they are related to chemical safety issues that are within NRC's regulatory jurisdiction) under the "Regulatory Component" and the "Safety Component" which are discussed in Section 2.1.1.3 and Figure 2.2. 52.2. Demonstrate that the Chemical Safety Function has sufficient training and practical chemical safety experience.	10 CFR 70.22(a)(6)	No change made to the License Application. Section 2.1.1.3 (c) further details the responsibilities for the different safety functions of the regulatory component, including chemical safety. Chemical Safety responsibilities are part of the Occupational Health and Safety program. To specifically address RAI 52.1, see Section 2.1.1.3 (c) for a description of these responsibilities. In reference to RAI 52.2, the training and qualification requirements for the chemical safety function are described in Section 2.1.1.3(e). Additional information is detailed in Section 3.4.2.2. At the CFFF, there is a training and qualification program for EH&S personnel. Training records are maintained and are available onsite for NRC review and inspection.
RAI 53 Describe the elements of training (e.g. appropriate personal protective equipment, lessons learned from conducting ISA) for the safe handling of hazardous chemicals within NRC's jurisdiction. Discuss the process in place to assure that the safety	10 CFR 70.22(a)(8)	No change made to the License Application. The ISA is continuously evaluated to ensure that Operating Experience and plant changes are accurately reflected. This process is iterative and reflective to assure that these insights are flowed down into plant procedures and training.

insights obtained from the ISA are incorporated into procedures and training. Discuss the types of personnel (e.g., operators, area supervisors, contractors, management) gaining such insights. Section 7.1.2.4 of the license application states that employees using hazardous chemicals are specifically trained in procedures for safe handling and disposal of them.		During site access training, all badged personnel receive hazard communication training discussing regulatory requirements associated with chemical safety, including safety data sheet locations and overview, procedural adherence, hazard identification, and labeling. Following this classroom training, plant personnel receive annual computer based training, including a test on Industrial Safety and Chemical Safety requirements. Personnel are trained to perform work activities in accordance with plant procedures. Training records are maintained and are available for NRC review and inspection. Handling of hazardous chemicals is not allowed if an individual is not current on his/her procedure or training requirements.
RAI 54 Clarify the chemical safety commitments related to compliance with 10 CFR 70 Subpart H described in Chapter 4.0 and Chapter 7.0. Section 7.1 states that all CFFF chemical safety commitments related to compliance with 10 CFR 70 Subpart H requirements are described in Chapter 4.0 of this license application. The purpose and scope of Chapter 7.0 is unclear.	10 CFR 70.62(a)	No change made to the License Application. The purpose of the Chemical Safety Program described in Chapter 7 is to assure hazards associated with the risk posed by chemicals used at the CFFF are evaluated, and that appropriate programmatic measures are taken to assure operations are performed in a safe manner [Section 7.1.2.2]. *The term "programmatic" is used here to refer to the organization, criteria, methods, and practices for conducting activities important to safety [NUREG 1520, rev 2].* Chemical Safety is an element of the Integrated Safety Analysis (ISA) Program described in Chapter 4.
RAI 55 Discuss the phases of operation (e.g., start up, shut down, maintenance, non-routine operations) addressed by the ISA. Clarify the phases of operation included in the ISA. Section 7.1 states that Chemical Safety is an element of the ISA program. The ISA is discussed in Chapter 4 of the license application, but the scope of the ISA is unclear as it relates to chemical safety. Section 4.1.1.2 discusses guidance for preparing the process description, but only discusses "normal operations".	10 CFR 70.62(a)	No change made to the License Application. The process description addresses each process that was analyzed as part of the ISA. This description also includes a discussion of the hazards (and interactions of hazards) for each process and the accident sequences that could result from such hazards, and for which the unmitigated consequences could exceed the performance requirements of 10 CFR 70.61. [Section 4.1.2.1 (c)] This includes all phases of operation, i.e. start up, shut down, maintenance, and non-routine operations. Also see response to RAI 53. Section 4.1.1.2 was removed in the revised Chapter 4 of the license application.

RAI 56 Discuss the relevant chemical hazards exposures included in the Chemical Safety Analysis (e.g. inhalation, dermal absorption, ocular contact, ingestion). Describe the method used to identify relevant exposure pathways that could lead to high or intermediate consequences as defined in 70.61.	10 CFR 70.62(a), 10 CFR 70.62(c)	No change made to the License Application. The Chemical Safety and ISA Programs evaluate all exposure pathways to assure worker and public safety. Consequences of unmitigated accident sequences are evaluated, and if the consequence could exceed the 10CFR70.61 performance requirements for a chemical under NRC jurisdiction as defined in Attachment 1, then a likelihood analysis is performed. Attachment 1 will replace Table 4.2 in the Renewal Application and will be incorporated into the ISA.
RAI 57 Discuss the following: a. Explain the apparent discrepancies between the detailed descriptions (e.g. process safety information) in ISA Summaries describing non-inhalation exposures as consequences of concern and tables lacking the quantitative standard for those consequences. 57 .2 List the quantitative standards of chemical safety used to assess the consequences to an individual from acute chemical exposure analyzed in the ISA. …Both the ISA Summaries and ISA Handbook identifies inhalation standards, such as ERPG-3 and ERPG-2, taken as consequences of concern for high and intermediate consequences. Specific ISA Summary chapters and detailed descriptions of specific accident sequences (e.g., chemical release sequences for ADU vaporization system) identify other consequences of concern …	10 CFR 70.65(b)(7)	Quantitative criteria were added to Attachment 1 for chemical exposures to HF. This attachment will replace Table 4.2 in the Renewal Application. This resolves any discrepancies.
RAI 58 Provide the basis for excluding that 49 percent HF pipe inside the enclosed area of the HF spiking station. Provide assurance all credible chemical safety related accident sequences in this HF spiking station have been considered. Demonstrate that there are no chemical hazards of licensed material	10 CFR 70.62 (c)(ii) and (c)(iii)	The purpose of the HF spiking station is to mix a small amount of 49% HF with liquid Uranyl Nitrate (UN). The "HF spiked" UN is then used as a feed material to the conversion process for producing uranium dioxide powder. There are two lines supplying material to the HF spiking station mix tanks, T-1280 and T-1281. Of the two supply lines, one is from the UN bulk storage tank which contains SNM. The second supply line is from the HF bulk storage tank which does not contain SMN.

and facility hazards (e.g. releases from HF piping) that could affect the safety of licensed materials (including routine and non-routine operations)…. Section 7.1.3.4 states that the Chemical Safety Analysis is a comprehensive assessment of each component within a defined system. This section also states that the scope and content of a Chemical Safety Analysis are customized to reflect the particular characteristics and needs of the system being analyzed. To gain reasonable assurance that the ISA demonstrates the appropriately identified hazards delineated in 10 CFR 70.62(c)(1)(ii) – (iii), the NRC staff performed vertical slice reviews of the ISA Summary, to provide assurance that the ISA is adequately evaluating chemical hazards within NRC Jurisdiction. On page 117 of ISA 03, Westinghouse states that the supply piping contains 49 weight percent HF solution. On page 126 of ISA 03, Westinghouse states, "A 5 percent (maximum) HF solution co-mingled with SNM is contained in vessels T-1280, T-1281… and the piping and equipment (e.g. pumps) attached to these vessels. The supply piping to T-1280 and T-1281 is excluded since no SNM is contained in this piping."		A recently completed process hazard analysis (PHA) evaluated the entire HF system. This included the 49% HF in the bulk storage tank and supply line as well as the 5 wt % HF contained in the mix tanks with the "spiked UN." The 5 wt % HF portion contains SNM due to the intentional comingling of HF and UN in the spiking station mix tank. The PHA determined there is no credible pathway to get backflow of SNM from the spiking station mix tank into the 49% HF supply line based on the system design. Backflow of SNM into the 49% HF supply line and bulk storage tank is analyzed in a Criticality Safety Evaluation (CSE). This CSE credits IROFS ADUHFS-121 (2.5 inch diameter passive overflow in top of the spiking station mix tanks) and ADUHFS-903 (vacuum break in the vent line of the spiking station mix tanks). These independent IROFS are included in the existing ISA and prevent backflow of SNM from the spiking station mix tanks into the 49% HF supply line. Regarding chemical exposures, the potential impact of the 49% portion of the HF system on the 5wt % portion was evaluated in the recently completed HF PHA. The potential to get a higher concentration of HF (greater than 5 wt%) into the "spiked UN" was determined to be possible if there was a Programmable Logic Controller (PLC) program error or load cell malfunction. The current ISA determined that spilling more than 3.23 gallons of 5 wt % HF could result in a high consequence event. Being sprayed by HF could also result in a high consequence dermal or ocular exposure event. The current ISA classifies a dermal or ocular HF exposure as a high or intermediate consequence event based on the percentage of HF (up to 50 %), exposure time, and percent of body (or eyes) exposed. IROFS have been established to prevent / mitigate against a HF release incident (all exposure pathways) for the portion of the HF system containing 5 wt % HF. The recently completed PHA determined that existing IROFS that prevent / mitigate against a 5 wt. % HF release incident will equally prevent / mitigate a higher concentration release incident. The ISA text will be clarified to reflect this information in the January 2018 ISA update. Discussions between Westinghouse and NRC on this ISA accident sequence are ongoing which may result in additional changes in the January 2018 ISA update.

RAI 59 Discuss and demonstrate that the Fire Safety Function has sufficient training and practical fire safety experience in nuclear facilities. Section 2.1.1.3.e of the license application describes the qualifications of Regulatory Engineering Functions.	10 CFR 70.22(a) (6)	No change made to the License Application. The function of Fire Safety falls under the 2.1.1.3 "c" and "e" requirements per Figure 2.2, as a Safety Component of the CFFF. Additional information is detailed in Section 3.4.2.2. At the CFFF, there is a training and qualification program for EH&S personnel. Training records are maintained and are available onsite for NRC review and inspection. The current job description for the "Senior Fire Safety Engineer" details the essential functions/supporting responsibilities of the position such as: Education – (Minimum Required) Bachelor's Degree in any engineering discipline or life or physical sciences, or other EHS related subject matter. (Preferred) - Bachelor's Degree in life or physical sciences, engineering, or other EHS related subject matter. Experience - (Minimum Required) 2-5 years fire safety experience is required. (Preferred) - 5 years is preferred and at least 2 years with nuclear experience. KSA (Knowledge/Skills/Abilities) - (Minimum Required) Demonstrate knowledge in Fire Safety Management Systems, Risk Management processes, Audit/Assessment Processes, Training processes, and Fire prevention. Demonstrate knowledge and have work experience in Fire Safety Suppression and Detection systems. Communication and writing skills are required as well as the ability to influence and effectively partner with others is required. (Preferred) - Minimum requirements plus knowledge in Fire Safety Rated Barriers, Fire Safety Interlocks and Project management skills. Certifications - (Minimum Required) NFPA-CFPS-Certified Fire Protection Specialist. (Preferred) - NFPA-CFPS-Certified Fire Protection Specialist, NFPA-CFPE-Certified Fire Plan Examiner, NFPA-CFI-Certified Fire Inspector I, and OSHA 30 Hour.
RAI 61 Discuss, in detail, combustible/flammable liquids and gases, including the design, construction, installation, labeling, testing, operation, and maintenance of tanks, containers, and cabinets; the control of ignition sources in these areas; and the procedures for the dispensing, handling, transfer, and use.	10 CFR 70.22(a)(6), 10 CFR 70.62(a)	No change made to the License Application. Chapter 8 of the license outlines the structure of the Fire Safety Program. The CFFF maintains a robust Fire Safety Program for protection of the site. The primary purpose of this Fire Safety Program is to assure that the opportunity for fires in and about the facility is kept As Low As Reasonably Achievable (ALARA). Fire protection is achieved by combinations of fire protection measures and systems. Such measures and systems are designed and maintained in accordance with industry standards and prudent industry practices. The standards and practices most often consulted are those of the National Fire Protection Association (NFPA). Chapter 8 goes on to discuss the details of the program. The commitments in this chapter are implemented in lower-tier site procedures to ensure safety and compliance. These flow down

		procedures describes the specific requirements related to design, construction, installation, labeling, testing, operation, and maintenance of tanks, containers, and cabinets; the control of ignition sources in these areas; and the procedures for the dispensing, handling, transfer, and use.
RAI 62 Discuss, in detail, the determination of areas of "substantial combustible loading," as referenced in Section 8.1.5.1 of the license application. Include training, procedures, and any correlations with the Fire Hazards Analysis.	10 CFR 70.22(a) (6), 10 CFR 70.62(a), 10 CFR 70.61	No change made to the License Application. Our areas of substantial combustible loading are listed out and reviewed based on recommendations in our Fire Hazards Analysis document which is an input into the ISA. The FHA uses a conservative assessment of several items for each of the above areas such as Items Relied on for Safety (IROFS), Safety Significant Components (SSCs), Fire Hazards, Ignition sources, fire loading, design basis fire scenarios, fire-fighting impact to name a few. The regulatory basis that is applied to this document is the Nuclear Regulatory Commission fire protection regulation contained in Title 10 of the Code of Federal Regulations, Part 70 Domestic Licensing of Special Nuclear Material. The requirements outlined by 10CFR70.22 and 70.65 provide the basis for the FHA. It also focuses on providing information to assist in assessing compliance with 10CFR70.61, 70.62, and 70.64. It also assesses the risk from fire in the facility in relation to the existing fire protection program to ascertain whether the facility meets the objectives of NUREG 1520 and the guidance provided by NFPA 801, Standard for Fire Protection for Facilities Handling Radioactive Materials. The last several statements are actual statements from the last revision of the FHA. The FHA also used several NFPA codes when compiling the document.
RAI 63 Describe how air effluents are controlled. Specifically, describe how gaseous effluent controls ensure compliance with the dose constraint in 20.1101(d).	10 CFR 20 Subpart B, D, F, K, L, M; 10 CFR 70.22(a)(7)	No change made to the License Application. HEPA filtration is installed on systems with the potential to release radioactive materials. Each radiological stack is continuously sampled to ensure release concentrations are less than the action level. The ALARA goal and action levels are based on RG 4.16 and the effluent concentrations listed in appendix B of 10 CFR Part 20.
RAI 64 Discuss how stack sampling is performed, or commit to follow Regulatory Guide 4.16, Monitoring and Reporting Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities, Revision 2, December 2010; and Regulatory	10 CFR 20 Subpart B, D, F, K, L, M; 10 CFR 70.22(a)(7)	No change made to the License Application. Air exhausted from each stack is sampled near the point of release after fans, HEPA filters, and/or scrubber systems. A vacuum source draws effluent air through filter media which is collected and analyzed on a daily frequency. Each stack is sampled continuously.

Guide 4.20, Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other Than Power Reactors, Revision 1, April 2012. American National Standards Institute (ANSI) standard ANSI N42.18-1980, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactive Effluents" provides additional guidance for effluent monitoring.		

Attachment 1

Severity Ranking	Consequence Description		
	Workers	Offsite Public	Environment
High	• Radiological dose greater than or equal to 1 Sv (100 rem) • 400 mg soluble uranium intake or greater • Chemical exposure greater than or equal to ERPG-3 • Dermal Exposure to an HF solution of less than 50 weight percent HF but greater than 11 weight percent over 10 percent of the worker body surface area (1610 cm2) for more than 30 minutes without medical treatment • A criticality accident	• Radiological dose greater than or equal to 0.25 Sv (25 rem) • 30 mg soluble uranium intake or greater • Chemical exposure greater than or equal to ERPG-2	• None
Medium	• Radiological dose greater than or equal to 0.25 Sv (25 rem) but less than or equal to 1 Sv (100 rem) • 150 mg soluble uranium intake or greater • Chemical exposure greater than or equal to ERPG-2 but less than ERPG-3 • Dermal exposure to an HF solution of 11 weight percent HF or less but greater than or equal to 1 weight percent over 10 percent of the worker body surface area (1610 cm2) for more than 30 minutes without medical treatment • Ocular exposure to HF solution of greater than 1 weight percent for more than 30 minutes without medical treatment.	• Radiological dose greater than or equal to 0.05 Sv (5 rem) but less than or equal to 0.25 Sv (25 rem) • Chemical exposure greater than or equal to ERPG-1 but less than ERPG-2	• A 24-hour averaged Radioactive release outside the restricted area greater than 5,000 times Table 2 Appendix B of 10 CFR Part 20
Low	• Accidents with radiological and/or chemical exposures to workers less than those above.	• Accidents with radiological and/or chemical exposures to the public less than those above.	• Radioactive releases to the environment producing effects less than those specified above.

* SV = Sieverts; ERPG = Emergency Response Planning Guidelines

Enclosure 2

Westinghouse November 2016 RAI Responses

REQUEST FOR ADDITIONAL INFORMATION	REG BASIS	WESTINGHOUSE RESPONSE
C. Nuclear Criticality Safety (NCS)	This section contains the responses to RAIs relating to nuclear criticality safety. In addressing the RAIs on nuclear criticality safety, CFFF includes TABLE 1 (at the end of the RAI responses) summarizing the corresponding regulatory requirements, and the section of NUREG-1520, Chapter 5 compliance for the various sections of applicability to the CFFF NCS program, which is more fully described in Chapter 6 of the license application.	
RAI 4. Commit to the 2005 version of American National Standards Institute/American Nuclear Society (ANSI/ANS) standard ANSI/ANS-8.19, or justify using an older version of the standard. Section 6.1 of the license application (Ref. 1) states that the Nuclear Criticality Safety (NCS) Program meets the requirements of ANSI/ANS-8.19-1996 as pertains to organization and administration. Section 5.4.3.2 of Nuclear Regulatory Commission (NRC) guidance (Ref. 2), states that the license application should contain justification if committing to other than the most current version of a standard endorsed by the NRC. Regulatory Guide 3.71 endorses the 2005 version of ANSI/ANS-8.19.	10 CFR 70.61(d)	The difference between the 1996 version of ANSI/ANS-8.19 and the current 2005 version is the (1) reference to follow the guidance for use of CAAS from the newer/reaffirmed version of ANSI/ANS-8.3-1997; R2003, and (2) reference to follow the guidance for emergency planning and response of ANSI/ANS-8.23-1997. CFFF is following the guidance of ANSI/ANS-8.3-1997, as evident from SNM-1107, Section 6.1.8, which states: "The CAAS radiation monitoring detectors are located to pursue conformance to the guidance of ANSI/ANS-8.3(1997) (as modified by Regulatory Guide 3.71), and compliance with 10CFR70.24." Furthermore, CFFF mostly complies with ANSI/ANS-8.23-1997. Section 9.0 of SNM-1107 has been revised as follows: "This program complies with the requirements of ANSI/ANS-8.23(1997) for nuclear criticality accident emergency planning and response ~~with the exception that CFFF shall comply with Section 8.3 evacuation drill requirements on a biennial frequency~~." Therefore, CFFF in its license application follows the guidance of the 2005 version of ANSI/ANS-8.19 and is justified in referencing ANSI/ANS-8.19-1996. However, for the sake of clarity going forward, CFFF has referenced the 2005 version of ANSI/ANS-8.19. In addition, CFFF will increase the frequency of evacuation drills to an annual basis to be in complete compliance with ANSI/ANS-8.23-1997; R2012.
RAI 5. Clarify use of the word "configuration" in the first paragraph of Section 6.1.1 of the license application (Ref. 1), which lists mass, moderation, and "configuration" as examples of controlled parameters. "Configuration" is not included in the list of parameters in Section 6.1.3 of the license application (Ref. 1), nor is this a normally recognized controlled parameter. This information is needed for clarity.	10 CFR 70.61(d)	It is recognized that "configuration", in this context, is a synonym of "shape". Therefore, the current wording in Section 6.1.3.4(1) "Geometry control is used to limit the shape, configuration or volume of SNM within specific process operations and vessels…" has been revised as follows: "Geometry control is used to limit the shape or volume of SNM within specific process operations and vessels..."

RAI 6. Section 6.1.1 of the license application (Ref. 1) states, "The defense consists of the bounding assumptions, criticality safety limits, and criticality safety constraints that, as a set, are uniquely sufficient to maintain the minimum subcritical margin against an initiating event." Clarify the difference between criticality safety "limits" and "constraints". Clarify what is meant by "uniquely sufficient to maintain the minimum subcritical margin against an initiating event." The minimum subcritical margin is an allowance for any unknown uncertainties in calculating Keff and is not typically associated with any particular initiating event or limits.	10 CFR 70.61(d)	It is agreed that the word "constraints" is not properly used, and should be changed to "controls." In addition, the statement "uniquely sufficient to maintain the minimum subcritical margin" is vague and non-descriptive. Therefore, the current paragraph in Section 6.1.1 that reads: "The defense consists of the bounding assumptions, criticality safety limits, and criticality safety constraints that, as a set, are uniquely sufficient to maintain the minimum subcritical margin against an initiating event" has been changed to: "The defense consists of the bounding assumptions, criticality safety limits and controls that, as a set, are uniquely sufficient to maintain subcriticality during normal and credible abnormal conditions."
RAI 7. Explain the difference between "audits" and "compliance audits" in relation to ensuring the reliability of administrative controls the third paragraph of Section 6.1.1 in the license application (Ref. 1). Section 5.4.3.2 of NRC guidance (Ref. 2) states expectations for various types of audits and assessments. Various terms are used throughout the nuclear fuel industry, and it is therefore necessary the terms be clearly understood.	10 CFR 70.61(d)	Section 6.1.9 in the license application describes the various types of audits performed by CFFF. Regarding ensuring the reliability of administrative controls, these are investigated during a compliance/process audit. Therefore, the word "audit" has been removed from the sentence in the third paragraph of Section 6.1.1, and the sentence has been revised as follows: "The reliability and effectiveness of administrative controls are assured through procedure reviews, training, experience, and compliance/process audits (as described in Section 6.1.9)."
RAI 8. In Section 6.1.3 of the license application (Ref. 1), commit that when using a single NCS control to maintain the values of two or more parameters, this constitutes only one component necessary to meet double contingency. By Section 5.4.3.2 of NRC guidance (Ref. 2), double contingency requires that at least two changes in process conditions are necessary for criticality, and that those changes in process conditions be independent. Though double contingency is not required for existing facilities by the rule, double contingency is both a commonly practiced and effective means to limit the risk of a nuclear criticality accident.	10 CFR 70.61(d)	For clarification, the following sentence has been added after the last sentence in the first paragraph of Section 6.1.1: "The use of a single NCS control to maintain the values of two or more controlled parameters constitutes only one component necessary to meet double contingency protection."

RAI 9. In Section 6.1.3 of the license application (Ref. 1), commit to when the control of parameters is based on measurement, the instrumentation used will be subject to facility management measures. Though Section 5.4.3.2 NRC guidance (Ref. 2) mentions this criterion when applied to specific controlled parameters (e.g., mass, density, enrichment), the concept applies to any parameter where control relies on measurement.	10 CFR 70.61(d)	The following sentence has been added to Section 6.1.1, third paragraph: "Any instrumentation relied upon to either verify or impose an NCS control or parameter is subject to CFFF management measures programs to assure the reliability of its intended function."
RAI 10. Section 6.1.3.1(2) of the license application (Ref. 1) states that an evaluation will be done to determine the controls necessary to prevent reaching the safety limit. Define the term "safety limit". The term "safety limit" is used in Section 6.1.3.1(2) of the license application, but not elsewhere in the chapter. Clear and unambiguous terms are necessary to communicate and implement nuclear safety concepts.	10 CFR 70.61(d)	Safety limit refers to the margin of safety for normal and credible conditions as outlined in Section 6.1.5.2, Limits of keff. For clarification, the sentence has been revised as follows: "The evaluation also considers normal operations and expected process upsets to determine the operating mass limit and the controls necessary to maintain subcriticality."
RAI 11. In Section 6.1.3.2 of the license application (Ref. 1), commit to evaluate the effect of fire suppressants and firefighting activities in areas subject to moderation control. By Section 5.4.3.2 NRC guidance (Ref. 2), the use of moderating fire suppressants can challenge moderation and possibly other controls and should be evaluated.	10 CFR 70.61(d)	No change made to the License Application. In Section 6.1.3.2 (4), CFFF commits to invoke ANSI/ANS-8.22 for moderator controlled areas. The requirement to evaluate the effect of fire suppressants and firefighting activities is invoked by (1) ANSI/ANS-8.22 Section 4.1.6, which requires moderator control requirements in fire-fighting procedures, and (2) ANSI/ANS-8.22 Section 4.2.10, which requires process evaluations (i.e., CSEs) to address the need for special controls for fire prevention and suppression activities.
RAI 12. Define the term "interstitial moderator" in Section 6.1.3.2(3) of the license application (Ref. 1). Commonly, the term is used to refer to the density of water filling the space between fissionable units in an array or other collection of units, but the term has also been used to refer to moderator that is intimately mixed with fissionable material. Clear and unambiguous terms are necessary to communicate and implement nuclear	10 CFR 70.61(d)	The term "interstitial" moderator is referring to the water that can potentially intrude into the analyzed system (i.e., interspersed moderation between fissile units) and considers the "full range" of densities from humidity and mist conditions to full water density. To avoid any unambiguity, Section 6.1.3.2(3)has been revised as follows: "Moderation controls (IROFS) are established to ensure that the interstitial moderator, or the water between fissile units, is maintained within the analyzed system's documented limits, for normal operation and expected process upsets. The most

safety concepts.		reactive credible "full range" densities (i.e., humidity/mist conditions to full water density) for interstitial moderator are modeled."
RAI 13. In Section 6.1.3.2(4) and (5) of the license application (Ref. 1), clarify the statement that Westinghouse will follow the "guidelines" of ANSI/ANS-8.22-1997. State whether "guidelines" consist of the requirements of the standard, its recommendations, or both. By Section 5.4.3.2 of NRC guidance (Ref. 2), when an applicant intends to conduct activities to which an NRC-endorsed standard applies, the application should contain a commitment to follow the requirements ("shall" statements) of the standard. The term "guidelines" is vague and does not make clear to what sections the licensee is committing. Clear and unambiguous terms are necessary to communicate and implement nuclear safety concepts.	10 CFR 70.61(d)	CFFF has changed the word "guidelines" in Section(s) 6.1.3.2(4) & (5) to "requirements."
RAI 14. In Section 6.1.3.2(4) of the license application (Ref. 1), clarify whether the bulleted commitments apply whenever moderation control is used, or only when moderation is the sole controlled parameter. The subject commitments are sub-bullets under the paragraph that starts "When moderation control is used as the sole controlled parameter…", but appear appropriate to moderation control generally. Clear and unambiguous terms are necessary to communicate and implement nuclear safety concepts.	10 CFR 70.61(d)	No change made to the License Application. The wording of the text implicitly implies that a "moderator control area" consists of the bulleted list in Section 6.1.3.2(4) as a requirement, as well as ANSI/ANS-8.22 (1997), and when moderation control is in the singular use as a controlled parameter.
RAI 15. In Section 6.1.3.3(3) of the license application, commit that all physical and chemical mechanisms that can affect concentration so as to challenge a concentration control limit will be considered and documented in Criticality Safety Evaluations (CSEs), or justify that the list of phenomena mentioned (e.g., precipitation, evaporation, freezing) is sufficiently all-inclusive.	10 CFR 70.61(d)	For clarification, Section 6.1.3.3(3) has been revised as follows: "The determination of concentration limits and controls will consider all physical and chemical mechanisms that can affect concentration such as precipitation, evaporation, freezing, settling, heterogeneity and chemical phase change events as appropriate."
RAI 16. In Section 6.1.3.3 of the license application (Ref.	10 CFR	The following sentence has been added to Section 6.1.3.3 as a new bullet:

1), commit that when using tanks containing concentration-controlled solution, the tank will be closed and locked to prevent unauthorized access. By Section 5.4.3.2 of NRC guidance (Ref. 2), all credible abnormal conditions must be considered to ensure that precipitating agents are not inadvertently introduced.	70.61(d)	"When using a tank containing concentration-controlled solution, the tank will be closed and locked to prevent unauthorized access."
RAI 17. Section 6.1.3.4(5) of the license application (Ref. 1) states, "Geometry controls will be maintained through management measures that include procedure reviews, training, experience, and audits." Section 6.1.3.10 of the license application states, "Spacing controls will be maintained through management measures that include procedure reviews, training, experience, and audits." Explain what is meant by these statements. Section 5.4.3.1 of NRC guidance (Ref. 2) states that applicants should commit to the double contingency principle, which requires that at least two changes in process conditions are necessary for criticality, and that those changes in process conditions be unlikely. Management measures are applied to controls to ensure that their failure is unlikely. However, management measures listed do not appear appropriate to passive geometry or spacing controls.	10 CFR 70.61(d)	No change made to the License Application. The function and effectiveness of all nuclear criticality safety-significant controls at CFFF, which includes both engineered and administrative controls, are verified and audited (per RA-108) to ensure proper performance. When geometry controls are relied upon for criticality safety, limiting equipment dimensions (e.g., cylinder radius) are maintained/verified through management measures (e.g., audits). The same applies when interaction is controlled by spacing items bearing fissile material (i.e., the amount of spacing/interacting requirement are verified through management measures). These controls are determined through criticality safety analysis of the normal and credible abnormal process upset conditions for the operation in question, which includes a double contingency analysis. In other words, the passive geometry or spacing controls are not single controls relied upon. The management measures are put in place to prevent degradation or loss of controls by verifying their intended functions. In addition, there are a few geometry and spacing controls that have an administrative component to their safety function. For instance, there are 3 sizes of polypaks available in the plant (7.5-, 8- and 9-inch). Operators must be trained on selecting the correct size polypak as well as what is the approved material loading for that polypak. Also, at various portions of the operation, polypaks will be hand-carried by an operator and the operator will need to know the spacing requirements for that polypak. Training and procedures are relied upon so that operations personnel will be able to readily recognize or detect failure of the control and take appropriate response actions for geometry and spacing control violations.
RAI 18. In Section 6.1.3.6(2) of the license application (Ref. 1), explain the phrase, "control of enrichment to less than the licensed limit…." State whether enrichment limits lower than the licensed limit will be used for criticality control. If so, state the controls that will be	10 CFR 70.61(d)	The maximum U-235 enrichment at CFFF is 5.0 wt%. Therefore, the basis/bounding assumption for performing CSEs is that all uranium is analyzed at 5.0 wt% U-235 enrichment, as this represents the maximum operating, or licensed, enrichment limit. No attempts are made at taking credit for lower enriched material in NCS space.

used to ensure limiting enrichments will not be exceeded. Section 5.4.3.2 of NRC guidance (Ref. 2) states that when enrichment is controlled, either a method of segregating enrichments is used to ensure different enrichments are not interchanged, or the most limiting enrichment is applied to all materials.		Enrichment values lower than the licensed limit are not used for criticality control. For clarity, the first sentence of Section 6.1.3.6(2) of the license application has been revised as follows: "Control of enrichment to not exceed the licensed limit is used to limit the percent of U-235 in a process, vessel, or container."
RAI 19. Section 6.1.3.7(2) of the license application (Ref. 1) states, "Nuclear criticality safety calculations have demonstrated that for particle sizes ≤ 150 microns in diameter, the material can be considered homogeneous." Provide technical justification for this assertion. Section 5.4.3.2 of NRC guidance (Ref. 2) states that heterogeneous effects should be considered whenever relevant. The technical basis for the subject statement is needed to determine if this is an adequate criterion for when heterogeneous effects should be considered.	10 CFR 70.61(d)	No change made to the License Application. CFFF has evaluated the impact of solid particles of UO_2 for heterogeneous systems to establish subcritical single parameter limits for heterogeneous solid uranium oxide particles (CN-CRI-06-7, Rev. 0). Various geometries were evaluated under optimal moderation with varying reflector conditions/material to establish the optimal particle radius (i.e., bounding from a reactivity standpoint). The evaluation shows that for systems where the particle size is greater than or equal to 1000 microns, the system is purely heterogeneous.
RAI 20. Provide technical justification for the exceptions to ANSI/ANS-8.5-1996 stated in Section 6.1.3.8(2) of the license application (Ref. 1), especially given the statement in the standard that Raschig rings should not be used in basic solutions unless chemical and physical limits have been determined and documented, due to the known corrosion of borosilicate glass in basic environments. Section 5.4.3.2 of NRC guidance (Ref. 2) states that if the applicant intends to conduct activities to which an NRC-endorsed standard applies, the application should contain a commitment to follow the requirements ("shall" statements) of the standard, subject to exceptions as discussed in Regulatory Guide 3.71. The technical basis for these additional exceptions needs to be understood.	10 CFR 70.61(d)	No change made to the License Application. The following technical justification for the use of Raschig rings in basic environments was taken from CSE-3-M, Section 4.1.1. "Regarding chemical degradation, the glass of the Raschig rings is at risk due to the harsh, strongly-basic chemical environment in the Q-tanks. In addition, the presence of fluorides in ammonium compounds provides a means by which hydrofluoric acid (HF), a known and effective glass dissolver, may be released in the tank(s) to structurally degrade the ring pack. Over the 30+ years of Q-tank operation the same ring pack has been in place and has not shown substantial degradation, providing empirical evidence that the basic solution environment, coupled with periodic acid-washing of the tanks, does not significantly affect the rings. Glass corrosion by aqueous ammonium hydroxide solutions is temperature dependent, with higher temperatures driving more rapid corrosion [10]. Solutions entering the Q-tanks do not come from sources at elevated temperatures; the temperature of the solution in the tanks is kept below 60° C / 140° F (SNM-1107 license condition [8]). The temperature of the wastewater solution in the Q-tanks is constantly monitored by the process control system and an alarm sounds if

		the temperature exceeds 130º F. To ensure that the license condition is met, the temperature is observed and recorded at least once per day (SSC ADUQTNK-108). In the event that the temperature of the wastewater is elevated above the maximum allowed, it is at least unlikely that the condition is not detected over multiple days such that significant ring degradation occurs. To prevent HF acid dissolution of the glass rings, the chemistry of the Q-tanks is maintained at a pH ⍰ 7 (SNM-1107 license condition [8]) at all times (except during acid washing) which maintains the fluorine in the ammonium compounds and not as free fluoride ions. Some limited acid additions are likely to have occurred over the operational life of the rings (feed mishaps, acid wash mishaps, etc.); however, dilution or other factors appear to have prevented any significant degradation of the rings by HF dissolution or other means. Prevention of inadvertent acid additions relies on valve alignments and management of isolation devices such as spool pieces and flange blinds. After acid washing a Q-tank system, spool pieces or blinds are required to be removed/installed on the acid feed lines to the Q-tanks (as applicable) prior to placing the tank system in operation (ADUQTNK-109)."
RAI 21. In Section 6.1.3.8(3) of the license application (Ref. 1), clarify whether the measurement of neutron absorbers includes verification of absorber dimensions in addition to composition.	10 CFR 70.61(d)	For clarification, the first bullet of Section 6.1.3.8 (3) has been revised as follows: "The absorber dimension and composition are measured, and documented in the applicable CSE, prior to first use."
RAI 22. Explain the significance of defining the terms "full reflection" and "partial reflection" as used in Section 6.1.3.9 of the license application (Ref. 1). These terms are not used elsewhere in the license application. Clarify if the definitions include that the 12-inches or 1-inch of water be "tight-fitting" and how these definitions apply in the presence of reflectors other than water (e.g., concrete).	10 CFR 70.61(d)	Reflection is one of the parameters that affect neutron multiplication, and this section states the acceptable ranges of reflection that are typically evaluated in the CSE. Section 6.1.3.9 has been revised as follows: Credible reflection conditions will be considered in the determination of all system limits and controls. The terms "full reflection" and "partial reflection" are defined as 12-inches and 1 inch of water equivalent (tight-fitting), respectively. If reflecting materials other than water are present (e.g. concrete), there reflecting properties will be evaluated for all credible conditions and justified, as appropriate. When less than full reflection is assumed, it shall be demonstrated that the reflection conditions modeled are the most reactive credible conditions, or appropriate controls (IROFS) will be established to maintain reflection within the applicable limits.

RAI 23. In Section 6.1.3.10 of the license application (Ref. 1), justify the first criterion for neutron isolation, specifically "units may be considered non-interacting when they are separated by a 12-foot air distance." Regulations require that all processes be shown to be subcritical under normal and credible abnormal conditions. Calculations performed to demonstrate subcriticality must therefore bound actual process conditions, including consideration for interaction between neighboring units. Neutron isolation may not be adequately ensured by a 12 foot air distance for sufficiently large units. A guideline often employed in the nuclear industry has been that single units may be considered isolated if separated by the "larger of 12-foot air distance or the greatest distance across an orthogonal projection of the largest fissile accumulations on a plane perpendicular to the line joining their centers." The criterion stated in Section 6.1.3.10 is deficient in this regard.	10 CFR 70.61(d)	No change made to the License Application. In CSEs, all the possible interactions between fissile units are taken into account and appropriate spacing controls are applied. The use of the 12-ft air distance is an industry standard that is used at CFFF when drawing system boundaries for analysis in large warehouse-size rooms. Secondly, this separation distance may also be used for upsets that are outside of the analysis in existing CSEs. For instance, in the event it was discovered that the contents of a container consisted of 6.0% U-235, then the 12-foot in air separation distance would be applied to that container to maintain subcriticality until additional analysis could be performed. The 12-foot in air distance would be applied to a unit that is already subcritical. No additional fissile units would be brought within 12 –foot in air to maintain subcriticality for that unit. The 12-foot distance isn't a normal operating limit. One of the largest, portable units at the CFFF with uranium is the bulk container which is limited to 1800kg UO2 powder. Numerous calculations have been performed to evaluate the interaction of the bulk container near other fissile units including common containers and process equipment. The calculations have shown that maintaining 12 inches of spacing between bulk containers and other SNM containers remains subcritical during normal and credible criticality upsets. Calculation CN-CRI-09-3 is provided as an example to demonstrate that 12 inches of separation distance maintains bulk containers and polypack storage racks units subcritical. . Considering the above information coupled with the fact that the mean free path of neutrons is on the order of magnitude ~1-10 cm, the 12-foot separation distance in air between fissile units is sufficient for CFFF operations for units to be considered non-interacting.
RAI 24. In Section 6.1.3.10 of the license application (Ref. 1), commit to having engineered controls, or where not feasible, augmented administrative controls that will be used for interaction control, and that their structural integrity will be sufficient for normal and credible abnormal conditions. By Section 5.4.3.2 of NRC guidance (Ref. 2), spacing upsets where spacing is only controlled administratively have commonly occurred.	10 CFR 70.61(d)	The following sentence has been added to the end of Section 6.1.3.10: "Engineering controls or where not feasible, augmented administrative controls will be used for spacing control. The structural integrity of the engineered controls (e.g., spacers or racks) should be sufficient for normal and credible abnormal conditions."
RAI 25. Section 6.1.4.2(1) of the license application (Ref.	10 CFR	For clarity, the second sentence in Section 6.1.4.2.(1) has been revised as follows:

1) states, "The evaluation identifies … the Safety Significant Controls necessary to ensure double contingency." Explain the statement. Define the term "Safety Significant Controls," how they are used, and whether they include administrative or only engineered controls. The term "Safety Significant Controls" is used in Section 6.1.4.2(1) of the license application, but is not defined anywhere else in the license application. Clear and unambiguous terms are necessary to communicate and implement nuclear safety concepts.	70.61(d)	"The evaluation identifies controlled parameters for the system, establishes bounding assumptions for other system parameters, and identifies the controls necessary to maintain subcriticality." Safety Significant controls (SSCs) at CFFF includes both engineered (passive and active) and administrative controls that provide basic protection to prevent any accidents that could impact health, safety, and the environment. IROFS are a subset of SSCs that are relied on to prevent potential accidents at a facility that could exceed the performance requirements in 10 CFR 70.61 or to mitigate their potential consequences. Controls credited in the fault trees established for credible criticality scenarios are designated as IROFS. The attributes that are credited in the subcritical by geometry determination are also designated as IROFS.
RAI 26. In Section 6.1.4.2(8) of the license application (Ref. 1), clarify whether CSEs must be performed by qualified NCS staff. Section 6.1.4.2(8) states that CSEs must be reviewed by a qualified Criticality Safety Technical Reviewer, but makes no mention of who performs and documents the CSEs. Similarly, Section 6.1.6 of the license application refers to a qualified Criticality Safety Technical Reviewer, but only in terms of performing independent verification of the CSEs. Similarly, Section 6.1.6 refers to a qualified Criticality Safety Technical Reviewer, but only in terms of performing independent verification of the CSEs. Organizational positions, functional responsibilities, experience, and qualifications of NCS personnel are necessary attributes of nuclear criticality safety. Section 5.4.3.2 of NRC guidance (Ref. 2) states that the applicant should meet the criteria in Section 2.4 of the same guidance, as it relates to the organizational positions, functional responsibilities.	10 CFR 70.61(d)	Only a qualified NCS staff member can perform a CSE. Section 6.1.4.2(7) has been revised for clarification as follows: "CSEs are performed by qualified NCS staff in accordance with guidelines provided in the CFFF procedure for CSE generation."
RAI 27. In Section 6.1.7 of the license application (Ref. 1), commit to provide distinctive NCS postings in areas,	10 CFR 70.61(d)	No change made to the License Application.

operations, work stations, and storage locations relying on administrative controls. By Section 5.4.3.2 of NRC guidance (Ref. 2), distinctive NCS postings ensure the operators understand the criticality safety significance of controls in their areas.		CFFF has committed to following the requirements of ANSI/ANS-8.1-1998 which has requirements in Section 4.14 on posting of administrative requirements. CFFF has implemented the requirement. The NCS postings at CFFF are distinct in that they have the same format and they are laminated on pink paper. ANSI/ANS-8.1-1998 Section 4.1.4 invokes the requirement for having NCS postings.
RAI 28. Section 6.1.8 of the license application (Ref. 1) states, "The CAAS [Criticality Accident Alarm System] radiation monitoring detectors are located to pursue conformance to the guidance of ANSI/ANS-8.3(1997)…." Clarify the statement. By Section 5.4.3.2 of NRC guidance (Ref. 2), states that if the applicant intends to conduct activities to which an NRC-endorsed standard applies, the application should contain a commitment to follow the requirements ("shall" statements) of the standard. The term "pursue conformance" is vague and does not make clear to what provisions in the standard the licensee is committing.	10 CFR 70.24(a)	The first sentence of the second paragraph of Section 6.1.8 has been revised as follows: "The CAAS [Criticality Accident Alarm System] radiation monitoring detectors are located in accordance with the requirements of ANSI/ANS-8.3(1997) (as modified by Regulatory Guide 3.71), and compliance with 10CFR70.24."
RAI 29. In Section 6.1.8 of the license application (Ref. 1), commit that the criticality accident alarm system (CAAS) will be designed to remain operational during credible events. By Section 5.4.3.1 of the NRC guidance (Ref. 2), a CAAS should be designed to remain operational during credible events such as a seismic shock equivalent to the site-specific, design-basis earthquake or equivalent value as specified by the Uniform Building Code, and during events such as fires, explosions, a corrosive atmosphere, and other credible conditions.	10 CFR 70.24(a)	In response to RAI 29, the following sentence has been added to the end of the second paragraph of Section 6.1.8: "The CAAS is designed to remain operational during credible events."
RAI 30. In Section 6.1.8 of the license application, commit to having a criticality alarm that is clearly audible in areas to be evacuated or to provide alternative notification methods documented effective in notifying personnel that evacuation is necessary.	10 CFR 70.24(a)	The following sentence has been added at the end of the last paragraph of Section 6.1.8: "The CAAS is clearly audible in all areas to be evacuated to ensure timely notification and evacuation or provide alternative notification methods documented effective in

By Section 5.4.3.1 of NRC guidance (Ref. 2), the purpose of the alarm is to initiate timely evacuation.		notifying personnel that evacuation is necessary."
RAI 31. In Section 6.1.8 of the license application (Ref. 1), commit to having fixed and personnel accident dosimeters in areas requiring a CAAS, and that they will be readily available to personnel responding to an emergency, with a method for prompt onsite dosimeter readout. By Section 5.4.3.1 of NRC guidance (Ref. 2), fixed and personnel accident dosimeters in areas requiring a CAAS ensure and protect response personnel from the consequences of a nuclear criticality accident. Such dosimeters ensure that response personnel are protected from the consequences of a criticality.	10 CFR 70.24(a)(3)	The following sentence has been added at the end of the last paragraph of Section 6.1.8: "Furthermore, areas where CAAS is deployed, CFFF provides fixed and personnel accident dosimeters for responding emergency personnel. Prompt onsite dosimeter readout is available in a location outside the immediate evacuation zone to protect response personnel from the consequences of a nuclear criticality accident."
RAI 32. Section 6.1.9 of the license application (Ref. 1) states, "… audits and assessments address the guidelines of ANSI/ANS-8-19(1996)." Clarify the statement. Confirm if it is the intent that audits and assessments will be done in accordance with the requirements of ANSI/ANS-8.19-1996. Section 5.4.3.2 of NRC guidance (Ref. 2) states that if the applicant intends to conduct activities to which an NRC-endorsed standard applies, the application should contain a commitment to follow the requirements ("shall" statements) of the standard. The term "guidelines" is vague.	10 CFR 70.22(a)(8)	Section 6.1.9 has been updated to reflect that CFFF will follow the requirements as well as the most recent version of the updated standard ANSI/ANS-8.19-2014. Section 6.1.9 has been revised as follows: "These audits and assessments address the requirements of ANSI/ANS-8-19-2014 and are performed as described in Chapter 3.0, Section 3.6 of this License Application."
RAI 33. Justify the triennial NCS program audit frequency in Section 6.1.9 of the license application (Ref. 1). Section 5.4.3.2 of NRC guidance (Ref. 2) states that all operating SNM process areas should be reviewed at some specified frequency, which depends on such factors as the complexity of the process, degree of process monitoring, and degree of reliance on administrative controls. A graded approach may be used to justify an alternative	10 CFR 70.22(a)(8)	The NCS program is audited by an independent party on a triennial basis. See the responses to RAI's 35 and 49 for additional information and justification. In addition to the triennial program audit, with this license renewal, the CFFF commits to following the requirement of an annual review to ascertain that procedures are being followed and that process conditions have not been altered, per the requirements in ANSI/ANS-8.19 and ANSI/ANS-8.1.

schedule. Section 6.1.9 of the license application states, "Program audits schedules are developed annually, with the complete NCS program assessed on a triennial frequency." No reasons are given for the triennial frequency.		
RAI 34. Justify the 5-year frequency of NCS compliance audits in Section 6.1.9 of the license application (Ref. 1). Section 5.4.3.2 of NRC guidance (Ref. 2), states that all operating SNM process areas should be reviewed at some specified frequency, which depends on such factors as the complexity of the process, degree of process monitoring, and degree of reliance on administrative controls. A graded approach may be used to justify an alternative schedule. Section 6.1.9 of the license application (Ref. 1) states, "Formal compliance audit schedules are developed annually, with one fifth of the fissile material processing areas described in the ISA audited annually, so that the complete set of operations making up the CFFF Integrated Safety Analysis (ISA) are assessed on a five year frequency." The assessments described in Section 6.1.9 have different frequencies, but the difference between them is not clear.	10 CFR 70.22(a)(8)	See response to RAI 35.
RAI 35. Describe the difference between the 5-year program assessments, described as "compliance audits that evaluate implementation of NCS requirements" and quarterly or semiannual facility walkthrough assessments, described as having "a focus on field compliance with established NCS controls" in Section 6.1.9 of the license application (Ref. 1). State how Westinghouse distinguishes between "higher risk" (requiring quarterly assessments) and "lower risk" (requiring semiannual assessments) operations. Section 5.4.3.2 of NRC guidance (Ref. 2) contains acceptance criteria for various types of audits and assessments. These assessments described in Section 6.1.9 of the license application have different frequencies, but the difference between them is not clear.	10 CFR 70.22(a)(8)	CFFF performs many types of audits. The NCS group performs the facility walkthrough assessments described in Section 6.1.9. Also, an annual review will be performed going forward with the license renewal per the requirements in ANSI/ANS-8.19 and ANSI/ANS-8.1. The frequency of facility walkthrough assessments is based on the risk of the system as determined by the applicable CSEs and ISAs. Specifically, the following criteria are employed to determine if a given process will be reviewed semiannually or quarterly: • Systems with no credible criticality scenarios: **Semiannually** • Systems with credible criticality scenarios with frequencies ≤ 1E-05 per year: **Semiannually** • Systems with credible criticality scenarios with frequencies > 1E-05 per year: **Quarterly** Systems that would otherwise qualify for a semiannual frequency may be assigned a quarterly frequency based on special circumstances (e.g., prior findings), as

		determined by the EH&S Engineering Manager. In addition, the NCS program is audited by an independent party on a triennial basis, and the ISA's are audited on a five year frequency as per the audit program described in Section 3.4. See the responses to RAI 49 for additional information.
RAI 36. In Section 6.1.10 of the license application (Ref. 1) clarify what is meant by stating that the combined process for procedures, training, and qualification "meets the guidelines of ANSI/ANS-8.19(1996) and ANSI/ANS-8.20(1991)." Confirm if it is the intent of Westinghouse that this process will satisfy the requirements of ANSI/ANS-8.19-1996 and ANSI/ANS-8.20-1991. Section 5.4.3.2 of NRC guidance (Ref. 2) states that if the applicant intends to conduct activities to which an NRC-endorsed standard applies, the application should contain a commitment to follow the requirements ("shall" statements) of the standard. The term "guidelines" is vague.	10 CFR 70.61(d)	It is the intent of CFFF to meet the requirements of ANSI/ANS-8.19-1996 and ANSI/ANS-8.20-1991. Therefore, the second sentence in Section 6.1.10 of the license application has been revised as follows: "This process is described in Chapter 3.0, Section 3.4 of this License Application, and meets the requirements of ANSI/ANS-8.19(1996) and ANSI/ANS-8.20(1991), as they relate to training, procedures, and the requirement that no single, inadvertent departure from a procedure could cause an inadvertent criticality."
RAI 37. Provide minimum qualifications for qualified NCS staff, including those who will perform and document CSEs and perform other NCS Program functions, and for qualified NCS Technical Reviewers who will perform independent verification. Describe the various positions related to NCS and their duties and minimum qualifications. Section 11.4.3.3 of NRC guidance (Ref. 2) states that the application should contain commitments regarding personnel qualification for managers, supervisors, technical staff, and others who perform regulated activities.	10 CFR 70.61(d)	The general NCS engineer qualification (RAF-125-5) consists of four areas of applicability which must be met to be considered a qualified NCS engineer at CFFF. The four areas where the NCS engineer must demonstrate proficiency are: A. Education/Experience – Hold a minimum of a Baccalaureate degree in science or engineering and two years of experience in the nuclear industry. B. Required Reading/Knowledge – Possess a working knowledge of the ANSI/ANS criticality safety standards (8-series) as well as CFFF related NCS/ISA manuals and procedures. In addition, required reading includes the textbook by Mr. Ron Knief on "Nuclear Criticality Safety Theory and Practice." Furthermore, NCS engineers must author three mentored CSEs as well as three mentored calculations. C. Off-site Criticality Safety Training – Complete a university or national laboratory sponsored NCS short/training course or equivalent education or job experience. D. General NCS Proficiency – Demonstrate knowledge of varied NCS topics such as Monte Carlo code usage, double contingency principle, techniques for demonstrating

		favorable geometry, preparing and implementing calc notes and criticality safety evaluation. Completion of the above training topics and satisfaction of RA-125 experience requirements represents the minimum requirements for the NCS Engineer Qualification at CFFF. For Senior NCS Engineer Qualification, additional requirements must be met including demonstrated proficiency in RA-310 requirements and completion of three mentored technical reviews. Additional requirements for both the NCS Engineer and Senior NCS Engineer pertain to process qualification for a specific process area at CFFF. Currently, the NCS criticality group at CFFF consists of four qualified individuals. They perform the following functions and primary duties. 1. NCS Engineer performs general nuclear criticality safety functions including authoring CSEs and calculations and provides support to operations, such as procedure reviews and performs facility walkthrough assessments. 2. Senior NCS Engineer performs the same duties as the NCS engineer in addition to performing technical reviews on CSEs and criticality safety calculations. 3. NCS Group Manager performs administrative duties along with general nuclear criticality safety functions including authoring and reviewing CSEs and facility walkthrough assessments.
RAI 38. Commit to follow the requirements of ANSI/ANS-8.23-1997, in regard to emergency response as related to NCS. Section 5.4.3.1 of NRC guidance (Ref. 2) contains this acceptance criterion. This is needed to ensure personnel are protected from the consequences of criticality.	10 CFR 70.24(a)(3)	The following sentence has been added at the end of Section 6.1 as follows: "Also, CFFF is committed to following the requirements of ANSI/ANS-8.23-1997 with regards to emergency response as related to NCS to ensure personnel are protected from the consequences of a criticality accident."
RAI 39. Commit to require personnel to perform activities in accordance with written, approved procedures, and that unless a specific procedure deals with the situation,	10 CFR 70.22(a)(8)	A second paragraph has been added in Section 6.1 as follows: "All activities that may affect NCS shall be performed in accordance with written and

personnel shall take no action until NCS has evaluated the situation and provided guidance. Commit to require personnel to report defective NCS conditions to the NCS Program. Section 5.4.3.2 of NRC guidance (Ref. 2) contains these acceptance criteria. They are needed to ensure an adequate response to off-normal conditions.		approved procedures. Should no specific procedure exist applicable to the situation, work shall not be initiated until such time that NCS staff has evaluated the situation and provided guidance. Furthermore, CFFF personnel shall report any defective NCS conditions to the NCS staff."
RAI 40. State whether density is relied on as a controlled parameter, and if so, commit that when process variables can affect the assumed density, the process variables are identified as controls. Section 5.4.3.2 of NRC guidance (Ref. 2) contains acceptance criteria for the use of density as a controlled parameter.	10 CFR 70.61(d)	A new subsection 6.1.3.11 has been added for density as follows: "Density is not relied upon as a controlled parameter. As concentration is a controlled parameter, density is only an implicit controlled parameter."
RAI 41. Section 6.1.2 of the license application (Ref. 1) states, "The relative effectiveness and reliability of NCS controls are considered during the CSE process." Describe what is meant by "relative effectiveness" and state that the effectiveness and reliability of NCS controls will be justified in the CSE. One of the main issues during an event (Ref. 3) with the S-1030 scrubber that controls did not work because they were based on invalid assumptions. Much effort is typically put into showing that controls are "reliable" by appealing to the type of control (e.g., passive, active) and describing management measures. An often overlooked consideration is ensuring that controls actually work and can fulfill their safety functions. That requires a much more detailed kind of review.	10 CFR 70.61(d)	The sentence has been revised as follows: "The effectiveness and reliability of NCS controls are considered, justified, and documented in the CSE process."
RAI 42. Section 6.1.3(b) of the license application states that when less-than-optimum (worst case credible) conditions are assumed to a given parameter, the basis will be documented and justified in CSEs. In Section 6.1.3(c), state that the independent review of any assumptions, and the basis for their acceptance, will be documented.	10 CFR 70.61(d)	The following statement has been added after the first sentence of Section 6.1.3 (c): "Furthermore, the independent review of the assumptions, including the basis or rationale for their acceptance, is separately documented at least every three years." Note that assumptions will be assessed by the independent review/audit team at a minimum of every 3 years. Also, the FWA process, which is the annual NCS operational

Another main issue in the event of the S-1030 scrubber (Ref. 3) was that of unvalidated assumptions, several of which turned out to be false even though they were carried forward through several revisions of the CSE. They were subject to peer review, yet there are no firm criteria for performing the peer review and there is very little documentation of that review was conducted. Requiring an independent assessment of any assumptions would at least ensure that more than one person has had to think carefully about them.		review for a process, also verifies that the NCS bounding assumptions are still valid.
RAI 43. In Section 6.1.3.1 of the license application (Ref. 1), state that when mass limits are derived for material assuming a given weight percent of uranium, compliance will be verified by either weighing the material and ascribing the entire mass to uranium, or conducting physical measurements to establish the actual weight percent. State that process variables that can affect the weight percent of uranium are identified as controls. State that any material associated with a fissile process will be treated conservatively as having a high content of uranium until demonstrated otherwise. An issue in the event involving the S-1030 scrubber (Ref. 3) was the non-conservative assumption that the material from the S-1030 scrubber was of low uranium content. This assumption was found to be incorrect. If all the material had been assumed to be uranium until measurements showed otherwise, the material would have been handled in a conservative manner (e.g., not pushed into two corners of the S-1030 scrubber when cleaning the scrubber of deposits). The assumption was based on process conditions that were not controlled. These commitments are not included in the section on mass control, even though there are acceptance criteria in NRC guidance (Ref. 2). The specific acceptance criteria listed above are from Section 5.4.3.2 (for parameters such as density, but the same principle applies generally to other parameters, including mass).	10 CFR 70.61(d)	For clarification, the following statement has been added at the end of Section 6.1.3.1 (1): "Note that when mass limits are derived based on weight percent of uranium, compliance will be verified by either weighing the material and ascribing the entire mass to uranium, or conducting physical measurements to establish the actual weight percent. Furthermore, process variables that can affect the weight percent of uranium are identified as controls." In addition, subsection, 6.1.3.1 (5) has been revised as follows: "Any material associated with a fissile process will be treated conservatively as having a high content of uranium until demonstrated otherwise." It should be noted that uranium mass is treated as a worse case until the uranium mass is confirmed via measurements.

RAI 44. In Section 6.1.3.5(2) of the license application (Ref. 1), state that when credit is taken for process characteristics (e.g., the physical and chemical properties of a process and/or process materials), the bounding assumptions and limits are documented and justified in the applicable CSE. Section 6.1.3.5(2) requires that credit for process characteristics must be documented, but does not require that it be justified. This is taken from Section 5.4.3.2 of NRC guidance (Ref. 2), where it states that process variables that can affect parameters should be controlled.	10 CFR 70.61(d)	Section 6.1.3.5(2) has been revised as follows: "When credit is taken for process characteristics (e.g., the physical and chemical properties of a process and/or process materials), the bounding assumptions and limits are documented and justified in the applicable CSE."
RAI 45. Section 6.1.3.5(3) of the license application (Ref. 1) states, "Utilization of process and/or material characteristics as controls is based on known scientific principles, established physical properties or chemical reactions, and/or experimental data supported by CFFF operational history." Explain what is meant by "known scientific principles". Explain what is meant by "established physical properties" and "experimental data," and provide examples. State that such credit cannot be based on operating history alone. Explain in detail how Westinghouse meets the commitments in Section 6.1.3.5(3) for the S-1030 scrubber, including which of the methods (i.e., known scientific principles, established physical properties or chemical reactions, experimental data) are being relied on, how they are being relied on, and how they are supported by operating history.	10 CFR 70.61(d)	Known scientific principles and established physical properties, including experimental data, refers to criticality handbooks, standards (e.g., ANSI, ASTM, ISO), and chemical and physical properties as described in the CRC Handbook of Chemistry and Physics. For clarification, Section 6.1.3.5 (3) has been revised as follows: "Utilization of process and/or material characteristics as controls is based on known scientific principles, established physical properties or chemical reactions, in conjunction with experimental data supported by CFFF operational history." The S-1030 scrubber does not credit any specific scientific principles, established physical properties or chemical reactions, and experimental data, per se. Based on operational experience, the frequency of inspection and cleanout of the scrubber internals have been implemented (CSE-1-E, Rev.13). This is tied to the NCS control parameter of concentration, namely the uranium concentration within the scrubber liquid.
RAI 46. Explain what is meant by "most reactive credible conditions" as applied to reflection in Section 6.1.3.9 of the license application (Ref. 1). Provide an example of how this would be applied within an enclosed process, such as a glovebox or ventilation ductwork. RAI 22 asked for clarification on commitments related to reflection. When reflection is not controlled, standard industry practice is that it is represented by 1 foot of	10 CFR 70.61(d)	In criticality safety analysis, normal and credible abnormal reflection is considered. The possibility of full water reflection is considered when performing an analysis, unless more efficient reflector materials are present (e.g., concrete), in which case such materials are considered (also see response to RAI 22). Note, however, that there are times when full water reflection is not credible, such as moderation control areas with controls in place to prevent the introduction of moderators into the area. The analyst must then demonstrate that the less than full reflection model is the most reactive credible condition.

tight-fitting water or 2 feet of concrete. Section 6.1.3.9 of the license application (Ref. 1) goes beyond this in allowing a third possibility, namely demonstrating that "the reflection conditions modeled are the most reactive credible conditions." This needs to be justified, especially where models of material within the scrubber included 1 inch tight-fitting water, even though there is a large amount of water present in and around the material under normal conditions. Using less conservative reflection conditions can result in a significantly higher mass limit than if full (1 foot) reflection is modeled.		An example of how less than full reflection can be applied within an enclosed process is the ModCon areas, where the introduction of moderators are controlled by limiting and amount of moderator introduced to the area. For clarity, the last sentence of Section 6.1.3.9 has been revised as follows: "When less than full reflection is assumed, it shall be demonstrated that the reflection conditions modeled are the most reactive credible conditions and appropriate controls (IROFS) will be established to maintain reflection within the applicable limits."
RAI 47. Explain what is meant by "establishes bounding assumptions for…system parameters" in Section 6.1.4.2(1) of the license application (Ref. 1), in regard to the contents of CSEs. Clarify whether the word "establishes" means that assumptions will be documented, justified (consistent with the words in Section 6.1.3.5[3]), or something else. The use of unvalidated assumptions has been a key issue in the event involving the S-1030 scrubber (Ref. 3). The commitment to "establish" bounding assumptions in analysis needs clarification.	10 CFR 70.61(d)	In a CSE, bounding assumptions are defined through the criticality evaluation process. That is, they need to be documented and their basis justified. The second sentence of Section 6.1.4.2(1) has been revised as follows: "The evaluation identifies controlled parameters for the system and the Safety Significant Controls necessary to ensure double contingency. In addition, the basis for bounding assumptions for other system parameters are documented and justified. "
RAI 48. Section 6.1.4.2(8) of the license application (Ref. 1) states that the independent review of CSEs by a qualified NCS Technical Reviewer, and the justification for their conclusions, must be documented. The CSEs for the S-1030 scrubber (Ref. 3), and related CSEs reviewed as part of the extent-of condition review, were reviewed by multiple individuals over the course of several revisions, yet were based on assumptions that turned out to be invalid. A robust peer review should have caught at least some of these issues (which were subsequently identified both by the NRC inspectors and by the contractors hired to do an independent assessment). To ensure a more thorough review, it must be more than a mere checklist that is signed off. The peer	10 CFR 70.61(d)	No change made to the License Application. The CFFF has revised its triennial NCS program audit to require an independent review of the CSEs to ensure that assumptions are properly documented, input data is traceable, and that limits and controls are clearly established. Typically, there are only 2-3 major CSE Revisions per year at CFFF. Therefore, reviewing the assumptions every three years by an independent assessor is reasonable.

reviewer should have to document what was looked at and why it was acceptable, including looking at any assumptions (see language in Section 6.1.3).		
RAI 49. Describe what is looked at during the triennial NCS Program audits, and who performs them, in Section 6.1.9 of the license application (Ref. 1). Justify the independence of the auditors from the program, and the basis for the triennial frequency. State how audit findings will be resolved. Section 5.4.3.2 of NRC guidance (Ref. 2) states that NCS program audits should be conducted at least once every 2 years, whereas Section 6.1.9 of the license application (Ref. 1) commits to assessing the entire program every 3 years. It is unclear what is meant by "auditing the entire program". In addition, mention is made of internal and external audit findings, but it is not clear if this is a commitment to conducting internal audits, external audits, or both, or who performs them. NRC guidance (Ref. 2) also says that reviews and audits should be independent. As follow-up to the event involving the S-1030 scrubber (Ref. 3), Westinghouse hired external contractors, who identified a number of issues with plant CSEs similar to those found by inspectors. A periodic external review of facility CSEs would seem beneficial. Section 5.4.3.3.4 of NRC guidance (Ref. 2) states that weaknesses identified during audits should be referred to the corrective action program, which is responsible for promptly and effectively resolving them. Section 6.1.9 of the license application (Ref. 1) states that the results of audits are documented and maintained, but does not state that they will be put into the licensee's corrective action program.	10 CFR 70.61(d)	No change made to the License Application. The NCS Triennial program audit scope shall address the guidelines of ANSI/ANS-8.19 (1996) and include reviews of the effectiveness of the Periodic Criticality Safety Evaluation (CSE) Technical Review Program. The audit team leader shall be an independent, qualified auditor with a background in criticality safety. The team leader is approved by the WEC Quality Programs Director. At a minimum, the audit team consists of three members, whereof at least two shall have experience in criticality safety, and at least one of the auditors shall have experience performing criticality safety evaluation. Independence of the audit team is ensured by using auditors, external to the CFFF, who have previously not performed any in-house analysis/work for SNM operations at the CFFF. The ANSI/ANS standards recommend that NCS related audits and surveillance does not exceed a 3-year time period. It is recognized that NUREG-1520 recommends audits should be conducted once every 2 years. The CFFF will maintain the frequency of the NCS program audit to be performed triennially due to the limited number of major CSE Revisions per year (2-3) coupled with the fact that the NCS program requirements do not change much ,if at all, within the three year period. Audit findings are entered into the Corrective Action Process (CAP) for resolution. This is in accordance with NUREG-1520, Rev.1, Section 5.4.3.3.4. The CAP is described in Section 3.8 of the License Application.
RAI 50. In Section 4.1.3.2 of the license application (Ref. 1), state that all changes to operations involving SNM will be evaluated by NCS and the affected operations. If safety analysis is not required for the change, the justification for that determination will be documented	10 CFR 70.61(d)	For clarification, the following text has been added to Section 4.1.2.2: (after second sentence in first paragraph) "Before a new operation with fissionable material is begun, or before an existing

on the Configuration Change Control Form. This shall include evaluating whether the validity of any underlying assumptions is impacted by the proposed change. In other words, ensure that NCS reviews all changes to fissile material operations, and justify if an analysis is not needed. In the event involving the S-1030 scrubber (Ref. 3), changes were made that invalidated the assumptions and controls in the process's safety basis; the event ensured partly as a result of the cumulative effect of many such changes. Section 4.1.3.2 of the license application states, "All subsequent changes that might affect the Baseline ISA are reviewed by the same safety disciplines that were involved in preparation of the Baseline ISA." However, this does not clearly state whether NCS (or operations) will be involved in the review of all facility changes. Such reviews are often done by a checklist; the analyst should rather be required to document the basis when deciding that a more detailed safety review is not required.		operation is changed, it shall be documented that the process remains subcritical under both normal and credible abnormal process upsets." (at the end of the first paragraph) "In the event a criticality safety analysis is not required for the change, a justification is provided and documented. This shall include evaluating whether the validity of any underlying assumptions is impacted by the proposed change."

TABLE 1
Criticality Safety Programmatic Requirements Matrix

License Application Chapter & Title	10 CFR 70 Requirement	NUREG-1520, Rev. 1 Section
6.1 NCS Program Structure	70.22(a)(8), Appendix A	5.4.3.2, 5.4.3.4.7 (7)
6.1.1 General Control Program Practices	70.61(d), 70.64(a)	5.4.3.1
6.1.2 Control Methods	70.61(d)	5.4.3.4.2
6.1.3 Controlled Parameters	70.61(d)	5.4.3.1; 5.4.3.2
6.1.4 Criticality Safety Documentation	70.61(d)	2.4; 5.4.3.2
6.1.5 Analytical Methods	70.61(d)	5.4.3.4.1 5.4.3.4.4 5.4.3.4.6
6.1.6 Technical Review	70.61(d)	2.4; 5.4.3.2
6.1.7 Posting of Limits and Controls	70.61(d)	5.4.3.2
6.1.8 Criticality Accident Alarm System (CAAS)	70.24	5.4.3.1, 5.4.3.2, 5.4.3.4.3
6.1.9 Audits and Assessments	70.22(a)(8), 70.62(d)	5.4.3.2, 5.4.3.3.4, 11.4.3.5
6.1.10 Procedures, Training, and Qualifications	70.62(d) & 10CFR19	5.4.3.2, 11.4.3.3
9.0 Emergency Management Program (compliance with the requirements of ANSI/ANS-8.23-1997)	70.24(a)(3)	5.4.3.1