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ADJUDICATIONS STAFF

December 22, 2004

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U. S. Nuclear Regulatory Commission
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Attn: Docketing & Services Branch

Re: Nuclear Fuel Services, Inc. – Docket No. 70-143

Dear Mr. Julian:

Today, Nuclear Fuel Services, Inc. ("NFS") is filing with the Presiding Officer NFS's Written Presentation in Response to Intervenors' Written Legal and Evidentiary Presentation.

NFS, State of Franklin Group of the Sierra Club, et al., and the NRC have agreed that during the pendency of the NRC Staff's motion for a protective order, that no party will publicly release NFS's Written Presentation. Therefore, NFS requests that NFS's Written Presentation not be made publicly available pending resolution of the NRC Staff's motion.

If you have any questions, please contact me at (202) 663-8455.

Sincerely,

Timothy J V Walsh

Timothy J. V. Walsh

Enclosures

cc: Alan S. Rosenthal, Esq.
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Shelly D. Cole, Esq.

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SECY-02

December 22, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Presiding Officer

In the Matter of	)	
	)	Docket No. 70-143
NUCLEAR FUEL SERVICES, INC.	)	Special Nuclear Material
	)	License No. SNM-124
(Blended Low Enriched Uranium Project)	)	

NOTICE OF APPEARANCE

The undersigned, being an attorney at law in good standing admitted to practice before the courts of the Commonwealth of Virginia, hereby enters his appearance as counsel on behalf of Applicant Nuclear Fuel Services, Inc. in any proceeding related to the above-captioned matter.[1] Mr. Walsh requests that he be served by e-mail with all papers henceforth served by e-mail in this proceeding.

Respectfully submitted,

Timothy J V Walsh

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Dated: December 22, 2004

[1] Daryl M. Shapiro, Esq. and D. Sean Barnett also remain counsel for Applicant Nuclear Fuel Services, Inc.

December 22, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Presiding Officer

In the Matter of	)	
	)	Docket No. 70-143
NUCLEAR FUEL SERVICES, INC.	)	Special Nuclear Material
	)	License No. SNM-124
(Blended Low Enriched Uranium Project)	)	

NOTICE OF APPEARANCE

The undersigned, being an attorney at law in good standing admitted to practice before the courts of the District of Columbia, and various federal courts, hereby enters his appearance as counsel on behalf of Applicant Nuclear Fuel Services, Inc. in any proceeding related to the above-captioned matter.[1] Mr. Travieso-Diaz requests that he be served by e-mail with all papers henceforth served by e-mail in this proceeding.

Respectfully submitted,

Matias F. Travieso-Diaz

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December 22, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Presiding Officer

	)	
In the Matter of	)	Docket Nos. 70-143-MLA, 70-143-MLA-2,
	)	70-143-MLA-3
Nuclear Fuel Services, Inc.	)	ASLBP Nos. 02-803-04-MLA, 03-810-02-MLA,
	)	04-820-05-MLA
(Blended Low Enriched	)	
Uranium Project)	)	Special Nuclear Material License No. SNM-124

APPLICANT'S WRITTEN PRESENTATION IN RESPONSE TO INTERVENORS' WRITTEN LEGAL AND EVIDENTIARY PRESENTATION

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Dated: December 22, 2004

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December 22, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Presiding Officer

In the Matter of	)	Docket Nos. 70-143-MLA, 70-143-MLA-2, 70-143-MLA-3
Nuclear Fuel Services, Inc.	)	ASLBP Nos. 02-803-04-MLA, 03-810-02-MLA, 04-820-05-MLA
(Blended Low Enriched Uranium Project)	)	Special Nuclear Material License No. SNM-124

APPLICANT'S WRITTEN PRESENTATION IN RESPONSE TO INTERVENORS' WRITTEN LEGAL AND EVIDENTIARY PRESENTATION

Pursuant to the Presiding Officer's Order of December 9, 2004, Applicant Nuclear Fuel Services, Inc. ("NFS") provides this Written Presentation in response to the Intervenors' Written Presentation[1] dated October 14, 2004 ("Intervenors' Presentation").

The Nuclear Regulatory Commission ("NRC") Staff has fully met its National Environmental Policy Act ("NEPA") responsibilities and correctly issued a finding of no significant impact ("FONSI") for each of the three NFS Blended Low Enriched Uranium ("BLEU") Project license amendments. In issuing the three FONSIs, the NRC Staff fully evaluated the expected and potential environmental impacts specific to BLEU-Project activities in three Environmental Assessments ("EAs"). The NRC Staff environmental evaluations also considered previous NEPA reviews, namely, a Department of Energy ("DOE") Environmental Impact Statement ("EIS") and the EA prepared for NFS's license renewal, which evaluated many of the potential environmental impacts from

[1] Legal and Evidentiary Presentation by State of Franklin Group of the Sierra Club, Friends of the Nolichucky River Valley, Oak Ridge Environmental Peace Alliance, and Tennessee Environmental Council ["Intervenors"] Regarding U.S. Nuclear Regulatory Commission Staff's Failure to Comply with National Environmental Policy Act in Licensing the Proposed BLEU Project (Oct. 14, 2004).

NFS's BLEU Project. These reviews all concluded that there would be no significant environmental impacts from BLEU Project activities.

The issues raised by Intervenors' Presentation are rather narrow. Intervenors have essentially conceded the validity of the factual underpinnings of the NRC Staff's environmental assessments of the BLEU Project. Intervenors also do not contend that the NRC Staff's environmental reviews were improperly performed. Nor do they suggest that normal operations of the BLEU Project will have any significant impact on the environment. Intervenors' <u>sole</u> challenge to the NRC Staff's NEPA review is their assertion that the risk from potential accidents is such that it amounts to a significant environmental impact and thus requires the preparation of an EIS. This argument is fundamentally flawed, however, because it relies on a misapplication of information contained in the BLEU Project Integrated Safety Analyses ("ISAs"). Through this misapplication, Intervenors greatly overestimate the probabilities of BLEU Project accidents. Intervenors also greatly exaggerate the potential consequences from BLEU Project accidents. Thus, Intervenors overstate BLEU Project risk.

As the discussion below demonstrates, a correct interpretation of the accident analyses contained in the BLEU Project ISAs and consideration of the actual (very low) consequences of potential BLEU Project accidents confirm the NRC Staff's NEPA conclusion that the risk of accidents occurring at the facility is so low that no EIS is required. Thus, the NRC Staff appropriately issued a FONSI for each of the three BLEU Project license amendments. Intervenors' arguments to the contrary must be rejected.

I. BLEU PROJECT BACKGROUND

NFS has received three license amendments to its NRC Materials License, Special Nuclear Material ("SNM") License No. SNM-124, to support process operations associated with the portion of the DOE BLEU Project that is being performed at NFS's Erwin, Tennessee facilities. See 68 Fed. Reg. 74,653 (2003).[2] The BLEU Project is part of a DOE National Nuclear Security Administration ("NNSA") program to reduce stockpiles of surplus highly enriched uranium ("HEU") through re-use or disposal as radioactive waste.[3] Re-use of the HEU as low enriched uranium ("LEU") is the favored option of the NNSA program because it converts nuclear weapons grade material into a form unsuitable for weapons, allows the material to be used for peaceful purposes, and permits the recovery of the commercial value of the material. 1st EA at 1-3.

A. The First BLEU License Amendment

On February 28, 2002, NFS requested its first BLEU Project license amendment to authorize the storage of LEU-bearing materials at the Uranyl Nitrate Building ("UNB"). NRC granted the license amendment on July 7, 2003,[4] and the UNB was ultimately constructed at NFS's Erwin site.[5] The amendment authorizes NFS to store, at the UNB, LEU nitrate solutions prepared and shipped to NFS from DOE's Savannah River Site. 1st EA at 1-2. The UNB will also store solutions prepared at the NFS Site. Id. at 2-5. The LEU solutions will be stored in tanks within a diked area of the UNB. Id.

[2] Nuclear Fuel Services, Inc., Notice of Receipt of Amendment Request and Opportunity to Request a Hearing for Oxide Conversion Building and Effluent Processing Building in the Blended Low-Enriched Uranium Complex, 68 Fed. Reg. 74,653 (Dec. 24, 2003).

[3] U.S. Nuclear Regulatory Commission, Division of Fuel Cycle Safety and Safeguards, NMSS, Environmental Assessment for Proposed License Amendments to Special Nuclear Material License No. SNM-124 Regarding Downblending and Oxide Conversion of Surplus High-Enriched Uranium (June 2002) ("1st EA") at 1-3.

[4] NFS, Inc., Amendment 39 (TAC Nos. L31688, L31739, L31748) – To Authorize Uranyl Nitrate Building at the Blended Low-Enriched Uranium Complex and Possession Limit Increase (July 7, 2003) ("First BLEU License Amendment").

[5] Environmental Statements; Availability, etc.: Nuclear Fuel Services, Inc., Notice of docketing, etc., 67 Fed. Reg. 66,172 (Oct. 30, 2002).

B. The Second BLEU License Amendment

On October 11, 2002, NFS requested its second license amendment to authorize modification to its processing operations in its BLEU Preparation Facility ("BPF").[6] The second amendment was granted on January 13, 2004.[7] The amendment authorizes NFS to downblend HEU-aluminum alloy and HEU metal to low-enriched uranyl nitrate at the existing BPF at NFS's Erwin site. 1st EA at 1-2; see also 68 Fed. Reg. at 796. NFS is already authorized to handle HEU at the BPF. 1st EA at 1-2. Process equipment previously used at NFS's 200 Complex at the Erwin site has been relocated to an existing production area in NFS's Building 333, designated as the BPF. 1st EA at 2-1. Approximately 7.4 metric tons of HEU-aluminum alloy and 9.6 metric tons of HEU metal will be used to produce high-enriched uranyl nitrate solution, which will be downblended with uranyl nitrate solution produced from 211.7 metric tons of natural uranium oxide to yield low-enriched uranyl nitrate solution in 5,000 gallon batches. Id. That uranyl nitrate solution will be transferred to and stored at NFS's UNB. Id. at 1-2.

C. The Third BLEU License Amendment

On October 23, 2003, NFS requested its third license amendment to authorize special nuclear material processing operations in its Oxide Conversion Building ("OCB") and Effluent Processing Building ("EPB"). 68 Fed. Reg. 74,653; see also Nuclear Fuel Services, Inc. (Erwin, Tennessee), LBP-04-05, 59 NRC 186, 187 (2004), aff'd in applicable part, CLI-04-13, 59 NRC 244 (2004). The third amendment was granted on July 30, 2004.[8]

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[6] Nuclear Fuel Services, Inc., Notice of Receipt of Amendment Request and Opportunity to Request a Hearing, 68 Fed. Reg. 796 (Jan. 7, 2003).

[7] NFS Inc., Amendment 47 – To Authorize Operations in the Blended Low-Enriched Uranium Preparation Facility, To Approve Integrated Safety Analysis Summary for Existing Processes in BPF (TAC No. L31693), and to Approve Revised ISA Summary Schedule (TAC No. L31782) (Jan. 13, 2004) ("Second BLEU License Amendment").

[8] NFS Inc., Amendment 51 – To Authorize Operations in the Blended Low-Enriched Uranium Oxide Conversion Building and Effluent Processing Building (TAC L31791) (July 30, 2004) ("Third BLEU License Amendment").

The amendment authorizes NFS to convert low-enriched liquid uranyl nitrate solutions into solid uranium oxide powder at the OCB and to operate effluent processing facilities at the EPB. 1st EA at 1-3; see also 68 Fed. Reg. at 74,653. Low-enriched uranyl nitrate solution is converted to uranium oxide powder in the OCB using the Framatome ANP, Inc. process, which has been in use for over 20 years by Framatome ANP at its Richland, Washington plant. 1st EA at 2-5. In that process, the uranyl nitrate solution is mixed with ammonium hydroxide and water to produce ammonium diuranate solids. Id. The solids are then separated using a continuous centrifuge and cross filter. Id. The solids are next dried in a screw dryer and then calcined in a rotary kiln under a flow of steam and hydrogen to reduce the solids to uranium oxide powder (which is then shipped offsite for further processing). Id. at 2-5 to 2-7. The dilute stream from the centrifuge is passed through a filter and ion exchange columns to remove uranium, which is recycled to the oxide conversion process. Id. at 2-7. The stream is then sent to the EPB for further treatment. Id. In addition to oxide conversion in the OCB, NFS will also dissolve natural uranium trioxide powder in nitric acid to convert it into uranyl nitrate solution, which will then be shipped offsite for further processing. Id.

In the EPB, the liquid effluent from the OCB is treated. First, sodium hydroxide is added to the effluent and ammonia is recovered and returned to the oxide conversion process. Id. The remaining effluent, consisting primarily of sodium nitrate diluted in water, is fed to an evaporator, concentrated, and further processed into a solid waste for disposal. Id. The steam from the evaporator is condensed, collected in tanks, sampled for verification of compliance with NFS's pretreatment permit, and then discharged to the sanitary sewer. Id.

D. Procedural History

NFS's three BLEU Project license amendment requests were the subject of several hearing petitions, including a petition concerning each request from Intervenors.[9] See LBP-04-05, 59 NRC at 189. The Presiding Officer held in abeyance the resolution of the petitions concerning the first two amendments pending the receipt of petitions on the third amendment. Id. at 187; see also Nuclear Fuel Services, Inc. (Erwin, Tennessee), LBP-03-1, 57 NRC 9, 17 (2003). The Presiding Officer ultimately admitted Intervenors to a proceeding concerning all three license amendment requests. LBP-04-05, 59 NRC at 198-200. The hearing requests from the other petitioners were denied. Id. at 200.

Pursuant to the Presiding Officer's September 30, 2004 Order adjusting the schedule for filing written presentations, Intervenors filed their written presentation on October 14, 2004. Pursuant to the Presiding Officer's December 9, 2004 Order adjusting the schedule for filing written presentations, NFS hereby submits its Written Response to Intervenors' Presentation.

E. Summary of Applicant's Written Response

The NRC Staff justifiably concluded that BLEU Project activities by NFS would not have a significant adverse impact on the environment. The NRC Staff prepared an EA that examined the potential environmental impacts of the entire BLEU Project, and subsequently followed up this analysis with two additional EAs that considered up-to-date information provided to the NRC by NFS. In addition to its three site-specific EAs for the BLEU Project, the NRC Staff also considered two prior environmental reviews

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[9] Request for Hearing by Friends of the Nolichucky River Valley, State of Franklin Group of the Sierra Club, Oak Ridge Environmental Peace Alliance, and Tennessee Environmental Council, (Nov. 27, 2002) ("1st Req."); Second Request for Hearing by Friends of the Nolichucky River Valley, State of Franklin Group of the Sierra Club, Oak Ridge Environmental Peace Alliance, and Tennessee Environmental Council, (Feb. 6, 2003) ("2nd Req."); Third Request for Hearing by State of Franklin Group of the Sierra Club, Friends of the Nolichucky River Valley, Oak Ridge Environmental Peace Alliance, and Tennessee Environmental Council, Regarding Nuclear Fuel Services' Proposed BLEU Project (Feb. 2, 2004) ("3rd Req.").

performed regarding HEU downblending activities at the Erwin, TN site: (1) a programmatic EIS prepared by DOE for the entire HEU disposition program; and (2) an EA prepared by the NRC Staff for NFS's license renewal. Based on all of this information, the NRC Staff evaluated the expected and potential environmental impacts from the BLEU Project, found no significant environmental impacts, and issued a FONSI for each of the three requested license amendments. Through this thorough process, the NRC Staff fully discharged its NEPA responsibilities. It need go no further.

Intervenors do not assert that the expected environmental impacts from normal BLEU Project operations require the preparation of an EIS. Indeed, these impacts are *de minimis*. Nor do Intervenors challenge any of the factual support for the NRC Staff's environmental review, or the methodology used in the review. Intervenors' also identify no areas of disagreement with the analyses and data provided by NFS.[10] Thus, their only argument is that an EIS should have been prepared to address the environmental impacts of potential accidents.

Intervenors argue that information contained in NFS's BLEU Project ISA Summaries regarding accident risk demonstrates that the project meets the NRC's "quantitative" criteria for the preparation of an EIS. Intervenors' Presentation at 22-32. That argument, however, is based on a fundamental misapplication of information contained in the BLEU Project's ISA Summaries. Intervenors erroneously assert that the accident sequence likelihood indices contained therein represent accident probabilities,

[10] Intervenors state that they "do not concede that the factual representations made by NFS in its license amendment application are correct." Intervenors' Presentation at 23, n. 14. This attempt not to concede the accuracy of NFS's factual representations is lacking in substance or effect since neither this nor any other statement in Intervenors' Presentation asserts that the factual representations are in fact incorrect. This attempt is also bereft of legal or factual support. It should therefore be disregarded as surplusage.

and that those probabilities are not remote and speculative, so that an EIS is required for the BLEU Project. Intervenors' Presentation at 29-31. As will be discussed below, the ISA Summaries do not present quantitative probabilities for accident frequencies and cannot be interpreted to stand for probabilities that specified events will take place. Rather, these summaries provide bounding likelihood maxima intended to demonstrate that the risk posed by each postulated accident has been reduced below the level specified in the NRC safety regulations.

Intervenors also greatly overstate the consequences that could result from chemical or nuclear criticality accidents by implicitly comparing the potential consequences of BLEU Project accidents to those of potential spent fuel pool and reactor accidents. In combination, Intervenors' overestimation of potential accident probabilities and overstatement of the potential consequences of the accidents greatly exaggerate the risk presented by the BLEU Project. Therefore, Intervenors' overblown quantitative claims fail to undercut the NRC Staff's FONSIs.

Intervenors also assert that the BLEU Project meets certain "qualitative criteria" for establishing the significance of the environmental impacts of proposed actions, thus requiring preparation of an EIS. Intervenors' Presentation at 32-37. This argument also fails because it is merely an expression of subjective disagreement with the FONSIs issued by the NRC Staff. Indeed, Intervenors' "qualitative" arguments provide no more than an unsupported attack on the EAs, are factually without basis, and do not raise issues warranting the preparation of an EIS.

II. THE NRC STAFF FULLY MET ITS STATUTORY AND REGULATORY NEPA REQUIREMENTS

The NRC Staff complied with applicable law and Commission regulations in implementing NEPA. The NRC Staff's EAs for the BLEU Project activities at NFS adequately analyzed expected and potential environmental impacts and justifiably found none to be significant. Therefore, the NRC Staff appropriately issued a FONSI for each of the three BLEU license amendments.

A. Legal Requirements for NEPA Review

1. What the NRC Staff is required to do

The Federal courts have provided clear guidance on the manner in which federal agencies must discharge their obligations under NEPA. These standards are set forth below. In addition, the NRC has promulgated regulations that specify the extent of the NEPA analysis that must be performed by the NRC Staff when evaluating a proposed licensing action. The NRC Staff has fulfilled both its statutory and regulatory obligations.

a) Federal Court Guidance

For every major Federal action significantly affecting the quality of the human environment, NEPA requires that the Federal agency contemplating the action evaluate the environmental impact of the proposed action. 42 U.S.C. § 4332(C). Stated another way, NEPA requires an agency to take a "hard look" at environmental consequences of its proposed actions. See Robertson v. Methow Valley Citizens Council, 490 U.S. 332, 352 (1989). The agency's NEPA review is subject to a "rule of reason," requiring consideration only of a range of "reasonably foreseeable" environmental impacts. San

Luis Obispo Mothers for Peace v. NRC, 751 F.2d 1287, 1300-01 (D.C. Cir. 1984), rehearing en banc granted on other grounds, 760 F.2d 1320 (D.C. Cir. 1985), aff'd en banc, 789 F.2d 26 (D.C. Cir.), cert. denied 479 U.S. 923 (1986); Dubois v. Dep't of Agric., 102 F.3d 1273, 1286-87 (1st Cir. 1996).

When there is only a risk that an accident related to the proposed federal action might render that action environmentally significant, a different analysis is required than that for evaluating environmental consequences certain to result. City of New York v. Dep't of Transp., 715 F.2d 732, 746 (2d Cir. 1983), cert. denied, 465 U.S. 1055 (1984). If only a risk of significant environmental impact exists, the agency must undertake a risk assessment, which is an estimate of both the consequences that might occur as a result of the accident and the probability of the accident's occurrence. Id. The agency is to calculate the overall risk of environmental impact by estimating possible consequences and then discount them by the probability of their occurrence. Id. at 747. Such discounting is appropriate because NEPA does not require consideration of "remote and highly speculative consequences." San Luis Obispo, 751 F.2d at 1300. And it is entirely appropriate for an agency to conclude that a federal action poses no significant risk to the environment based on its calculation of overall low risk. See City of New York, 715 F.2d at 752.

b) NRC NEPA Regulations

The NRC's NEPA regulations are contained in 10 C.F.R. Part 51. For certain licensing actions, the NRC Staff must prepare an EIS. 10 C.F.R. § 51.20(b). For other licensing actions, a categorical exclusion applies, and no environmental review is required. 10 C.F.R. § 51.22. For all other licensing actions, Part 51 requires the preparation of an EA. See International Uranium (USA) Corporation (White Mesa Uranium Mill), LBP-02-19, 56 NRC 113, 122 (2002); 10 C.F.R. § 51.21.

At the outset of a licensing action that does not fit under the § 51.20 or § 51.22 criteria, the NRC Staff will begin its NEPA review by preparing an EA. 10 C.F.R. § 51.31; Pacific Gas & Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-877, 26 NRC 287, 290-91 (1987). If the NRC Staff concludes the project will not significantly impact the environment, it will issue a FONSI. An EIS is required if the NRC Staff's EA indicates that the action will significantly impact the environment. 10 C.F.R. § 51.20(b)(14).[11] In determining whether an EIS is required, "remote and speculative" impacts do not significantly affect the quality of the human environment for NEPA purposes and will not prompt preparation of an EIS by the NRC Staff. See San Luis Obispo, 751 F.2d at 1300.[12]

2. **EA Requirements**

An EA shall "identify the proposed action and include a 'brief' discussion of the need for that action, the alternatives to it, and the environmental impacts of the proposal and the alternatives." Diablo Canyon, 26 NRC at 290; 10 C.F.R. § 51.30. The scope of the NRC Staff's NEPA review of a license amendment is more limited than that performed for the issuance of the initial license. Florida Power & Light Company

[11] 10 C.F.R. § 51.20(b)(14) states, in applicable part: " (b) The following types of actions require an environmental impact statement or a supplement to an environmental impact statement: . . . (14) Any other action which the Commission determines is a major Commission action significantly affecting the quality of the human environment."

[12] In making its determination as to whether an EIS should be prepared, the Staff need not undertake a full probabilistic risk assessment analysis ("PRA"). Carolina Power & Light Co. (Shearon Harris Nuclear Power Plant), LBP-01-9, 53 NRC 239, 252, aff'd CLI-01-111, 53 NRC 370 (2001). Rather, the Staff's EA determination to issue a FONSI rather than prepare an EIS can be "based on existing materials available to it, probabilistic and otherwise, supplemented by additional information it might obtain from the Applicant in an environmental report or through requests for additional information." Id.

(Turkey Point Nuclear Generating Plant, Units 3 and 4), LBP-90-16, 31 NRC 509, 537 (1990). Thus, the NRC Staff's EA for the BLEU Project's license amendments needed to evaluate only those environmental impacts beyond those previously considered that would result from the proposed license amendment. Id.

In addition, in conducting its NEPA evaluation, the NRC Staff may rely on or adopt information from the NEPA review done by another federal agency. Philadelphia Electric Co. (Limerick Generating Station, Units 1 and 2), ALAB-785, 20 NRC 848, 868 n.65 (1984). Such reliance supports one of NEPA's central purposes, which is "to promote coordinated government action with an awareness of the long-range environmental implications of Government decisions." U.S. Energy Research & Dev. Admin. (Clinch River Breeder Reactor Plant), CLI-76-13, 4 NRC 67, 81 (1976). For example, where a federal agency has prepared a programmatic EIS, such long-range planning will be "useless if the merits of the plan must be reconsidered for each subsequent Federal action." Id. Therefore, the NRC Staff's NEPA evaluation can consider or integrate issues previously addressed in another federal agency's programmatic EIS to avoid duplication of analysis. Id. at 80. As will be discussed in later sections, the NRC Staff appropriately considered and relied on analyses contained in the DOE EIS and the License Renewal EA and, therefore, did not need to perform those analyses again.

3. Challenges to NRC Staff NEPA Review

The NRC Staff's determination of whether preparation of an EIS is required "may be made an issue in an adjudicatory proceeding." Consumers Power Company (Palisades Nuclear Plant), LBP-79-20, 10 NRC 108, 120 (1979) (citing Northern States Power

Company (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41 (1978).

However, a party who intervenes in a licensing proceeding to challenge the NRC Staff's NEPA evaluation has a significant burden to meet. It is not enough for such an intervenor to reference matters that ought to have been considered in the evaluation and, without anything more, demand repudiation of the NRC Staff's NEPA determination for failure to consider the referenced matters. Vt. Yankee Nuclear Power Corp. v. Natural Res. Def. Council, 435 U.S. 519, 553-54 (1978).

In a case such as the instant one, where the NRC Staff has prepared EAs and issued FONSIs, to challenge the NRC Staff's actions and the attendant determination that there is no need to prepare an EIS, "[a] Petitioner raising a NEPA claim is required to show a dispute exists between it and the applicant or the NRC Staff on a material issue of fact or law." Turkey Point, LBP-90-16, 31 NRC at 537 (citing 10 C.F.R. 2.714(b)(2)(iii)). The petitioner's arguments must establish that there exists a "decisive legal impediment to the issuance of the license amendment in issue; i.e., that that issuance was in direct violation" of NEPA. White Mesa, LBP-02-19, 56 NRC at 117. The petitioners must provide credible evidence either establishing that the proposed activities to be licensed might have a significant environmental impact or controverting the NRC Staff's claim that its review and FONSI determination were sufficient. Id. at 123. Failure to do so, or refutation by the NRC Staff or applicant of the petitioner's claim, will require the dismissal of the petition. Id. at 140-41.

B. The NRC Staff's EAs for the BLEU Project Justifiably Concluded that No Significant Environmental Impacts Would Result

The NRC Staff concluded that the BLEU Project's environmental impacts did not meet the level of significance that would prompt preparation of an EIS. The lack of significance finding was based on prior NEPA reviews of HEU disposition alternatives, NRC Staff examinations of NFS's ongoing operations at the Erwin site, and a comprehensive examination of the BLEU Project's specific impacts.

As part of its environmental review for its HEU disposition program, DOE analyzed in considerable detail the HEU downblending operation at the NFS Erwin, TN site.[13] In a separate action, the NRC Staff prepared an EA and issued a FONSI in 1999 following its evaluation of the environmental impacts of NFS's license renewal application. The NRC Staff's review evaluated many, if not most, of the processes involved in the BLEU Project. These earlier environmental reviews will be discussed in detail below.

For the BLEU license amendments, the NRC Staff NEPA review needed to evaluate only environmental impacts not previously analyzed that would result from BLEU Project activities. The NRC Staff could – and did – rely, where appropriate, on the programmatic analyses already performed by DOE for its HEU disposition program and the NRC Staff's review performed when NFS applied for a license renewal. The NRC Staff conducted a site-specific NEPA review for the BLEU Project and prepared three EAs. The NRC Staff concluded that the approvals of the BLEU license amendments were not "major Commission action[s] significantly affecting the quality of

[13] DOE/EIS-0240, Office of Fissile Materials, Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement (June 1996) ("DOE EIS").

the human environment." White Mesa, LBP-02-19, 56 NRC at 122-23. From that conclusion, completion of the NRC Staff's NEPA review required only the issuances of FONSIs. 10 C.F.R. § 51.31.

1. The DOE EIS considered a range of environmental impacts to the NFS site area that would or could potentially result from similar downblending operations

In June 1996, DOE issued an EIS to evaluate alternatives for the disposition of domestic origin, surplus HEU. DOE EIS at 1-1. DOE concluded that all reasonable alternatives for HEU disposition involved downblending the HEU material into LEU, either for future commercial use or to be disposed of as low-level radioactive waste. Id. at 1-4. The DOE EIS, therefore, assessed potential environmental impacts for the four sites where HEU conversion and blending could occur, including the NFS facility in Erwin. Id.[14] DOE considered the four sites "a range of reasonable alternatives." Id. at 2-36.

The DOE EIS assessed, for each of the four candidate sites, "the direct, indirect, and cumulative environmental consequences of the reasonable alternatives under consideration." DOE EIS at 2-12. It described all then-existing potentially affected environments based on available environmental documents and models. Id. According to the DOE EIS, each of the candidate sites reviewed and updated the corresponding environment descriptions presented in the EIS to ensure an accurate representation of the site and surrounding environment. Id.

[14] Two government sites were assessed, DOE's Y-12 Plant at the Oak Ridge Reservation in Oak Ridge, TN and DOE's Savannah River Site in Aiken, SC, as well as another commercial facility, the Babcock & Wilcox Naval Nuclear Fuel Division facility in Lynchburg, VA. Id.

The DOE EIS analyzed three blending technologies, along with the environmental impacts from the transportation of materials. DOE EIS at 1-4. The DOE EIS considered a No Action Alternative and four other alternatives, which represented different ratios of downblended end product, commercial use versus waste. Id. at 1-6. In addition, for two of the ratios (substantial commercial use and maximum commercial use), DOE also analyzed variations in the number and type of sites used in the downblending campaign. See id. at 4-118–120. DOE identified its Preferred Alternative as that which blended down the most HEU for resulting commercial use. Id. at 1-6.

The DOE EIS described the conversion and blending operations then ongoing at the NFS facility -- primarily the conversion of HEU into a product for the naval nuclear fuel program. DOE EIS at 2-34. The DOE EIS noted that the NFS facility is one of only two commercial sites in the U.S. capable of providing HEU processing services. Id. In addition, when NFS obtained a license amendment in 1993 authorizing it to downblend HEU, the NRC concluded, after conducting a Safety Evaluation Report ("SER") analysis, that no significant impact to health, safety, or the environment would result and a categorical exclusion under 10 C.F.R. § 51.22 applied. Id. at 2-35.[15] The DOE EIS further discussed the "complete environment, safety, and health program that includes all relevant areas" at the NFS facility. Id.

The DOE EIS also provided a detailed description of the affected environment for the NFS site, including the effects on land resources, site infrastructure, noise, water resources, geology and soils, biotic resources, cultural resources, public and occupational health, and waste management. DOE-EIS at 3-97–120. Chapter four of the DOE EIS

[15] The NRC did not prepare an EA for the 1993 license amendment authorizing HEU downblending because none was required. Id.

discussed environmental consequences for the alternatives. Id. at 4-1. That discussion included an assessment of potential public and occupational health risks presented by downblending HEU to LEU as uranyl nitrate, with further conversion into uranium oxide, based on accident scenarios that represented bounding cases. Id. at 4-22, 4-32–42, E-64–67. Based on those bounding cases, the DOE EIS discussed the potential releases of radioactivity and hazardous chemicals that could impact site workers and the offsite population. Id. at 4-32–40. The analyzed scenarios included a tornado, straight winds, an aircraft crash, a truck crash, nuclear criticality, process-related accidents, and an evaluation basis earthquake. Id. at 4-30, E-65. The accident analysis considered a downblending process similar to that which is being employed as part of the BLEU Project. See id. at 2-20–22, E-65 – 67.

The DOE EIS concluded that all four sites analyzed have the capacity to process HEU material with "minimal impacts to workers, the public, or the environment." DOE EIS at 2-36.

2. In the NFS License Renewal EA, the NRC Staff considered environmental impacts from continuing site operations, including potential impacts from accidents, and issued a FONSI

On February 4, 1999, the NRC published its EA ("License Renewal EA") and issued a FONSI for the license renewal of NFS's Special Nuclear Material License SNM-124 to authorize the continued processing of HEU for the U.S. Navy and processing of HEU scrap to recover uranium, among other support and decommissioning activities at the NFS facility in Erwin. 64 Fed. Reg. 5,681. The License Renewal EA discussed both the proposed license renewal alternative and a No-Action Alternative, i.e., not renewing the authorization for HEU processing. The License Renewal EA assessed both the environmental impacts from normal operations at the facility and those under accident conditions. Id. at 5,681-83.

The License Renewal EA discussed the expected radiological discharges to the atmosphere and surface water under normal operations. Id. at 5,681. Dose assessments were performed for radiological releases to the air and water. The total effective dose equivalent ("TEDE") for the maximally exposed individual was estimated at 2.7 mrem per year. Id. at 5,682. The NRC sets a TEDE limit of 100 mrem per year. Id. Thus, expected doses from normal operations at the NFS facility were well below the regulatory threshold. Id. The impact analysis considered the population living within 80 kilometers of the plant. Id. The total population dose of .4 per-Sv/yr was an insignificant addition to the background dose of 1000 per-Sv/yr for the affected population of 950,000 people.

The License Renewal EA also discussed non-radiological discharges to the air, surface water and groundwater. Id. at 5,682. Air emissions primarily consisted of volatile organic compounds, carbon monoxide, and nitrogen oxides; normal emissions were not expected to significantly impact the environment because estimated concentrations were two to three orders of magnitude less than the most stringent state standard. Id. Surface water was expected to be protected by enforcing limits and monitoring programs. Id. Although there were indications of previous groundwater contamination, NFS had undertaken several remediation efforts to address the contamination. Id.

In addition, the License Renewal EA also calculated potential fatalities from the transportation of waste to the Envirocare of Utah facility – which is a round trip travel distance of approximately 6560 kilometers. Id. at 5,682. The NRC Staff estimated 0.72 fatalities would occur over the course of 2874 shipments. Id.

According to the NRC Staff, no impacts were expected on land use, biota, or cultural resources. Id.

With regard to potential accidents, the License Renewal EA recognized that the activities performed at the facility could result in an "uncontrolled release of radioactive material to the environment." Id. at 5,682. Therefore, the NRC Staff conducted accident analyses based on three postulated representative accidents: 1) a drop of contaminated dirt during remediation activities; 2) the failure of a high efficiency particulate air filter as a consequence of fire; and (3) a generic criticality event. Id. For the first two events, the License Renewal EA concluded that potential exposures to the maximum exposed individuals were a small fraction of annual background radiation. Id. at 5,683. Regarding a potential nuclear criticality, potential exposures were higher. Id. The NRC Staff calculated the prompt, external, and internal doses that would result from an inadvertent criticality to be 0.5, 1.5, and 0.026 rem, respectively. However, the NRC Staff noted that two independent, concurrent failures must occur before a criticality event could occur, thus the possibility of such an event was considered extremely low. Id. Accordingly, the NRC Staff concluded the overall risk from a criticality accident was acceptable. Id.

Based on these analyses, the NRC Staff determined that the environmental impacts associated with the issuance of the license renewal amendment would not be significant. Id. As a result, the NRC Staff issued a FONSI. Id.

3. The NRC Staff's environmental review of the three BLEU Project license amendments extensively considered expected and potential environmental impacts and found that no significant environmental impacts would result

Through the site-specific EAs prepared for the BLEU Project, the NRC Staff evaluated actual and potential environmental impacts that would result from the BLEU Project's operations at NFS. The NRC Staff issued a FONSI for each license amendment. See 67 Fed. Reg. 45,555 (July 9, 2002); 68 Fed. Reg. 61,235 (Oct. 27, 2003); 69 Fed. Reg. 34,198 (June 18, 2004). Thus, the NRC Staff determined that granting the BLEU Project license amendments would not significantly affect the quality of the human environment at the NFS site. This finding has relieved the NRC Staff of any obligation to prepare an EIS. White Mesa, LBP-02-19, 56 NRC at 116 n.3; Diablo Canyon, ALAB-877, 26 NRC at 290.

a) The First EA and the First BLEU License Amendment FONSI

The BLEU Project is a limited addition to existing operations at NFS's Erwin site. 1st EA at 1-1. Because the NRC evaluated all environmental impacts for existing conditions and operations in the License Renewal EA, the NEPA evaluation for the BLEU Project evaluated only those impacts that would result from BLEU Project activities and any cumulative impacts on existing plant operations. Id.

To support the NRC's environmental review for the BLEU Project, NFS submitted environmental documentation for the three proposed license amendments. 1st EA at 1-1. To avoid segmentation of the environmental review, the NRC Staff prepared the 1st EA to evaluate the environmental impacts for the entire BLEU Project at NFS. Id. The NRC Staff noted that it would conduct an additional environmental review for each license amendment to determine if the 1st EA had adequately assessed the environmental effects. Id.[16] As will be discussed in the ensuing sections, the NRC Staff later prepared two additional EAs and issued a FONSI for each of the three license amendments.

The 1st EA described in specific detail the proposed processing operations for all of the activities involved in the BLEU Project. 1st EA at 2-1–2-6. It also described expected direct impacts to the environment in the form of air, water, and solid waste effluents. Id. at 2-8–2-12. The 1st EA also discussed NFS's radiation protection program and noted that NFS already maintains an NRC-approved Part 20 radiation protection program. Id. at 2-13. Indeed, the 1st EA highlighted the fact that NFS had considerable experience in uranium processing and had previously conducted HEU downblending at the site. Id. Accordingly, NFS had "not identified any unique radiological safety issues, associated with the proposed action, that would require significant changes to the existing radiological safety procedures." Id.

[16] The NRC Staff also conducted safety evaluations in connection with the proposed license amendments. The information evaluated in the NEPA and safety evaluations overlaps, yet there are significant differences between the two reviews' content and purposes. According to Staff guidance, "The NEPA document does not address accident scenarios, rather it addresses the environmental impacts which would result from the accident and is therefore dependent on certain information from the [safety evaluation report] ("SER"). Accident scenarios (i.e., frequency, probability) are addressed in the SER. Much of the information describing the affected environment is also applicable to the SER...and the NRC Staff should ensure consistency between the NEPA Document and the SER." NUREG-1748, Environmental Review Guidance for Licensing Actions Associated with NMSS Programs (August 2003) at 1-4 ("NUREG-1748").

For alternatives to the proposed action, the 1st EA considered the no-action alternative. Because other alternatives to the proposed action were considered in the DOE EIS, they were not reanalyzed by the NRC Staff in the 1st EA. 1st EA at 2-1. The no-action alternative was defined as not authorizing the requested license amendments. Id. at 2-14. According to the NRC Staff, the only impact from the no-action alternative would be the relocation of NFS's activities in support of the DOE HEU disposition program to other sites, resulting in no net environmental gain. Id.

Chapter Three of the 1st EA describes in considerable detail the affected environment for the NFS site. The NRC Staff set forth a detailed discussion of the site's (1) description; (2) climatology, winds, meteorology, and air quality; (3) demographic, socio-economic, and environmental justice concerns; (4) land impacts; (5) geology, mineral resources, and seismicity; (6) hydrology; (7) biota; (8) background radiological characteristics; and (9) the nature and extent of existing contamination at the site. 1st EA at 3-1–3-19.

The climatology and meteorology data provided in the License Renewal EA remained applicable in the 1st EA evaluation. Id. at 3-1–3-3. The 1st EA concluded there were no disproportionate adverse impacts on minority or low-income populations because the percentages for minority and low income populations surrounding the plant were similar to that in the rest of Tennessee. Id. at 3-5. The 1st EA also concluded no National Register or Historic places would be affected by the BLEU Project. Id. at 3-6. No wetlands would be disturbed by BLEU Project activities. While sóme wetlands were being eliminated from remediation efforts unrelated to the BLEU Project, that loss was more than offset by a larger wetland increase to another area of the NFS site. Id. at 3-6–

3-7. Specific earthquakes were not associated with known faults near the NFS site, and there was no evidence of geographically recent fault displacement. Id. at 3-7. One Federally Endangered mussel species was located near the NFS site, but that location was upstream of the NFS site, and therefore, no impact was expected to that species as a result of BLEU Complex Operations. Id. at 3-11.

With regard to prior contamination at the site, the NRC Staff concluded that decommissioning efforts would serve to limit the potential for contaminants to migrate beyond the facility's boundaries. Id. at 3-14. Any radiological hazard from construction related fugitive dust was expected to be low. Id. Although contaminants have been identified in surface waters near the NFS facility, the reported levels were below regulatory limits. Id. As for groundwater contamination, the NRC Staff noted that the impacts from existing contamination had already been evaluated in the License Renewal EA. Id. at 3-16. There were no plumes of contamination beneath the BLEU Project locations on the site, and the NRC Staff expected that the use of safety controls would significantly reduce the potential for loss of containment from BLEU Project activities that could further contaminate groundwater. Id. at 3-16.

Chapter Four of the 1st EA describes the effluent and environmental monitoring program NFS proposed for the BLEU Project so that it could evaluate potential public health impacts and ensure compliance with NRC requirements. Radiological and nonradiological airborne effluents from the BLEU Project will be treated and released through stacks. Air emissions from the BPF will be released through the NFS main stack, for which air monitoring was evaluated in the License Renewal EA. 1st EA at 4-1. Air emissions from the BLEU Complex will be discharged from new stacks. Id.

Nonradiological effluents will be monitored in accordance with state permits, and approval of the BLEU Project license amendments will be contingent on obtaining any required new or modified permits. Id.

Liquid effluents from the BPF will be treated at the waste water treatment facility ("WWTF"), while liquid effluents from the BLEU Complex will leave the site via the sanitary sewer to the publicly-owned treatment works. Id. at 4-4. Monitoring of the WWTF remains unchanged since the License Renewal EA. Id. The WWTF has previously treated effluent from HEU downblending operations, and the radiological and nonradiological constituents from the BLEU Project's downblending will be similar to previous operations. Id. Thus, no changes are expected for the BPF state permit.

BLEU Complex effluents will be monitored for their radiological and nonradiological constituents under a separate pre-treatment permit from the publicly owned treatment works. 1st EA at 4-4. Monitoring of current effluents remains unchanged from the License Renewal EA. Id. at 4-4.

NFS monitors impacts to the surrounding area by sampling ambient air, soil, vegetation, surface water, sediment and groundwater. 1st EA at 4-4. The License Renewal EA summarized the current monitoring program, and NFS will expand such monitoring to cover BLEU Project activities. Id. at 4-6.

Chapter Five of the 1st EA discusses the potential environmental consequences of the proposed action. The NRC Staff described expected non-radiological and radiological environmental impacts for normal operations, which consisted of the release of low levels of chemical and radioactive constituents to the atmosphere and surface water. 1st EA at 5-1 – 5-7. Operations from the BLEU Project are not expected to have a significant impact on non-radiological air quality. Id. at 5-1. Current emissions estimates, which include former operations, are expected to bound emissions from the BPF. Id. When BLEU Complex emissions are added to currently permitted emissions, all limits are met, except for nitrogen oxides; however, concentration limits at the nearest site boundary are two to three orders of magnitude less than the most stringent state air quality standards. Id. The NRC Staff expects the state regulators to set permit limits so that surrounding air quality is not adversely affected.

Surface water runoff flows towards the northwest boundary of the NFS site, and then into Martin creek. 1st EA at 5-2. Mitigation measures will be employed during construction to limit the impact of runoff to surface waters. Id. Surface water quality will be protected from future site activities by enforcing limits and monitoring programs. Id. Discharges to the Nolichucky River are not expected to have a significant impact on water quality because of dilution in the river. Id. Under normal operations, the BLEU Project will not discharge effluents to the groundwater. Id. at 5-3. No adverse land impacts are expected, nor are any expected to biotic resources or cultural resources. Id. at 5-3. As stated in the EA, all radiological and nonradiological risks to workers and the public from transportation were analyzed in the DOE EIS, and no significant impacts were identified. Id. at 5-4.

The NRC Staff evaluated potential environmental impacts under accident conditions, which consisted of higher amounts of chemical and radiological releases over a shorter period of time. 1st EA at 5-7–5-10. The 1st EA contains a detailed discussion of the potential impacts from accidents that could result from the BPF and BLEU Complex processing facilities and tank storage of the processing solutions. Id. at 5-7–5-10. The

EA recognized that accidents at the BLEU Project facilities "can potentially impact worker safety, public health and safety, and the environment." Id. at 5-7. The NRC Staff's "evaluation examine[d] the inventory of materials to be used, the processing parameters, and the reactions occurring in the process, to evaluate potential hazards in each facility." Id.

The NRC Staff concluded that BLEU Project processes would be safe and posed no significant risk to the environment. With regard to the BLEU Project's processing facilities, many of the process operations "are patterned after existing, NRC licensed processes." 1st EA at 5-7. Therefore the NRC Staff concluded that, in addition to the safety control information provided by NFS, "operational experience and history build confidence that operations can be executed safely." Id. The proposed process operations, such as downblending of HEU and HEU storage, were, indeed, "very similar" to corresponding processes already licensed at NFS. Id. at 5-8. Thus, the NRC Staff evaluated only those potential hazards associated with new operations. Id. NFS would employ concentration limits and favorable geometry process vessels to prevent inadvertent nuclear criticalities. Id. For chemical safety, NFS would treat process off gases through scrubbers and HEPA filters and maintain concentrations of hazardous chemicals to safe levels. Id. The NRC Staff concluded that "the safety controls to be employed in the processes for the BPF appear to be sufficient to ensure planned processing will be safe." Id.

The NRC Staff also analyzed potential hazards that could result from tank storage of process chemicals and concluded that safety controls to be used were sufficient to ensure that storage would be safe. Id. at 5-9. Those controls include concentration limits and safe geometry to prevent inadvertent nuclear criticality. For chemical safety, controls include tank berms for spill control and isolation, and equipping tanks with level controls for overfill protection. Id.

The NRC Staff then analyzed potential accidents from operations at the BLEU Complex, which consists of processing the LEU solution into uranium oxide powder in the OCB and treatment of the liquid effluent in the EPB. Id. at 5-9–10. Concentration limits and safe geometry are employed to prevent inadvertent criticalities. Id. at 5-10. For the LEU solution conversion to uranium oxide, NFS will use the Framatome ANP Inc. process, which was also already licensed by the NRC. Id. at 5-10. Thus, the NRC Staff had "additional confidence that oxide conversion can be operated safely." The NRC Staff determined that the activities not covered by the Framatome process include storage of uranyl nitrate previously discussed, and processing effluents at the EPB. Id. The NRC Staff considered the effluent processing to be a "common industrial process." Id. Hazards will be limited by the removal of uranium and ammonia from the process stream. Id. Ultimately, the NRC Staff reached the same conclusion for BLEU Complex operations, namely, that NFS will employ safety controls sufficient to ensure planned processing to be safe. Id.

The 1st EA also evaluated potential cumulative impacts from the proposed action by assessing those impacts that would add to known impacts from the existing facility. 1st EA at 5-10. While the NRC Staff expected some increase in chemical effluent, NFS would be required to comply with existing and new environmental permits set by state authorities to ensure any effluent increase is within acceptable limits. Id. at 5-11. In addition, the NRC Staff expected a "negligible" increase of radiological doses considered in relation to total facility doses. Id. Thus, the NRC Staff concluded that any additional impact from the BLEU Project operations would "represent a small change to existing conditions in the area surrounding the plant." Id.

In addition to the review of potential accident consequences in the EA, the NRC Staff stated that it would also review the detailed accident analyses contained in NFS's ISAs. Id. at 5-1. NRC Staff review of the ISAs would confirm compliance with the performance requirements of Part 70, which ensures that "all important accident scenarios and consequences are evaluated prior to a decision on the amendment requests." Id.[17]

On July 9, 2002, the NRC Staff issued a FONSI, supported by the evaluation contained in the 1st EA, for the First BLEU License Amendment to authorize construction and operation of the UNB. 67 Fed. Reg. 45,555. The FONSI summarized the environmental impact evaluation the NRC Staff conducted in the 1st EA, which concluded that normal operations would result in the release of small quantities of radioactive material and chemical effluents to the environment. Id. at 45,556. Under accident conditions, while the potential existed for higher concentrations of such materials to be released into the environment over a shorter period of time, the NRC Staff concluded that the safety controls to be employed were sufficient to ensure safe operations. Id. at 45,556-57. Thus, based on the evaluation conducted in the 1st EA, the NRC Staff concluded that the risk of an adverse environmental impact from construction

[17] As noted earlier, approval of each license amendment followed the preparation of a Safety Evaluation Report ("SER") by the Staff. Each SER analyzed the extensive accident analyses conducted by NFS and concluded that all necessary safety controls were in place to ensure safe operations at the BLEU Project's facilities. See UNB Amendment SER (July 2003) at 94; BLEU Preparation Facility SER (Jan. 2004) at 21.0-1; OCB/EPB SER (July 2004) at 68.

and operation of the UNB was so low that no significant environmental impact would result. Therefore, the NRC Staff issued the FONSI. Id. at 45,557.

b) The Second BLEU License Amendment EA and FONSI

On October 27, 2003, the NRC Staff published an EA ("2nd EA") and FONSI for the Second BLEU License Amendment. 68 Fed. Reg. 61,235. The 2nd EA analyzed the environmental impacts from the BPF and any cumulative impacts on existing plant operations. Id. at 61,236. The 2nd EA noted that impacts for existing conditions were evaluated in the License Renewal EA, and that impacts for the entire project were already evaluated in the 2002 1st EA. Id. The 2nd EA presented up-to-date information and analysis of the environmental impacts used by the NRC Staff to determine that it would issue a FONSI for the Second BLEU License Amendment to authorize the BPF. Id.

The 2nd EA summarized the five processes that would make up the BPF. 68 Fed. Reg. 61,236. The blending of natural uranium and HEU was previously authorized at NFS, and some of the process operations were previously assessed in the License Renewal EA; however, some of the processes were new and required the license amendment. Id.

Just as in the 1st EA, the 2nd EA noted that other alternatives to the proposed action were addressed in the DOE EIS and not re-analyzed in the 2nd EA; the only alternative available here was to deny the license amendment. Id. at 61,236. The 2nd EA also noted that a full description of the site and its characteristics had been given in the License Renewal EA and the 1st EA. Id. It also noted that a full description of the effluent monitoring program had already been provided in those prior two EAs. Id. In addition, a full description of the environmental impacts of the proposed action had already been provided in the License Renewal EA and the 1st EA. Id. at 61,237.

For normal operations at the BPF, the 2nd EA concluded that radiological effluents would be well below regulatory limits. 68 Fed. Reg. 61,237. For example, the total annual dose estimate for the maximally exposed individual from all planned effluents is less than 1 mrem, compared to NRC's annual public dose limit of 100 mrem and the air effluent limit of 10 mrem. Id. The NRC Staff expected no increase in doses to workers because the types and quantity of material, and the processing to be done, will be similar to what is already licensed at the site. Id. Nor was any significant change to surface water quality expected because of already enforced effluent limits and monitoring programs. There were no expected adverse impacts to groundwater, land, biotic resources, cultural resources, or adverse impacts from transportation activities. Id.

For potential accident scenarios, the NRC Staff concluded that information provided by NFS in its detailed ISA confirmed earlier information and conclusions that the safety controls to be employed would ensure that planned operations would be safe. Id. Thus, the risk presented by BPF operations was so low that the NRC issued a FONSI. Id. at 61238.

c) The Third BLEU License Amendment EA and FONSI

On June 18, 2004, the NRC Staff published an EA ("3rd EA") and FONSI for the Third BLEU License Amendment. 69 Fed. Reg. 34,198. The 3rd EA addressed operations at the OCB and EPB, and any cumulative impacts on existing operations. Id. at 34,199. Similar to the review scope of the 2nd EA, the third EA presented up-to-date information and analysis evaluating the environmental impacts of the Third BLEU License Amendment to authorize the OCB and EPB. Id. According to the NRC Staff, the proposed actions under the third license amendment were "consistent with the proposed actions[s] previously assessed in the [1st EA]." Id.

The 3rd EA summarized the nature of the proposed action, namely the four processes to be used in the OCB and the three processes to be used in the EPB. Id. The EA described one alternative to the project, the no action alternative, as other alternatives to the action were already analyzed in the DOE EIS. Id. The 3rd EA explained that the affected environment, the NFS effluent monitoring program, and a full description of the environmental impacts of the proposed action had already been evaluated in the License Renewal EA and the 1st EA. Id. at 34,200. The EA noted that, during its review of the amendment request, a different location was indicated for the stack constructed for the OCB than that presented in NFS's Supplemental Environmental Report. Id. The NRC agreed with NFS's conclusion that the correctly noted stack location would not change the result of the chemical and radiological consequence analysis. Id.

The 3rd EA stated that all radioactive effluents are monitored to ensure compliance with NRC regulations, and those monitoring reports are submitted on a semi-annual basis. Id. Effluents are also monitored for nonradiological constituents, and state authorities were expected to set limits that are protective of public health and safety. Id. NFS had also obtained a new sewer pre-treatment permit from the local utility. Id. Accordingly, the NRC Staff concluded that the environmental impacts from the Third BLEU License Amendment were fully consistent with those described in the 1st EA. Id.

The NRC Staff reviewed expected environmental impacts from normal operations and those under potential accident conditions. Id. With regard to accident conditions, the NRC Staff recognized that higher concentrations of chemical and radiological constituents could be released into the environment over a shorter period of time. Id. The NRC Staff had evaluated potential impacts from accidents in the 1st EA for activities authorized by the Third BLEU License Amendment, namely, processing the LEU solution into uranium oxide powder in the OCB and treatment of the liquid effluent in the EPB. Id. In addition, NFS had provided detailed accident analyses in its ISA for the Third Amendment. Based on its review of the ISA, the NRC Staff concluded that "potential accidents identified in the ISA are consistent with the previous evaluation" and that "the safety controls to be employed in the proposed action appear sufficient to ensure planned processing will be safe." Id. at 34,200-01.

With regard to cumulative impacts from the proposed action, the NRC Staff concluded, as it did in the 1st EA, that "[a]fter reviewing the updated information provided by NFS... the cumulative impacts represent an insignificant change to the existing conditions in the area surrounding the NFS site." Id. at 34,201. The cumulative impacts were previously described in the 1st EA, id., and consisted of a "possible" increase in chemical process effluent and a "negligible" increase in radiological effluent. 1st EA at 5-11. Thus, the risk presented by OCB and EPB operations was so low that the NRC issued a FONSI. 69 Fed. Reg. at 34,201.

III. INTERVENORS' ARGUMENTS ON WHY PREPARATION OF AN EIS IS REQUIRED MISINTERPRET THE FACTS AND MISAPPLY THE INTEGRATED SAFETY ANALYSES PREPARED PURSUANT TO 10 C.F.R. PART 70

As discussed in Section II, the NRC Staff conducted a comprehensive review of the BLEU Project's potential environmental impacts, including accident risks, building on the environmental reviews of the DOE HEU disposition program and of the license renewal for NFS's Erwin operations. The NRC Staff justifiably concluded that FONSIs were warranted for each of the BLEU Project license amendments. Intervenors raise quantitative arguments as to why the NRC Staff must prepare an EIS. Those arguments are based on a fundamental misapplication of accident likelihood information presented in the BLEU Project's ISA Summaries and a mischaracterization of potential BLEU Project accident consequences. In addition, Intervenors assert that the BLEU Project meets the NRC's "qualitative" criteria for determining that a Federal action significantly impacts the environment. This qualitative argument is nothing but an unsupported attack on the results of the NRC Staff's NEPA evaluations. Intervenors' errors in their presentation are fundamental in nature and fatal to their argument.

A. Intervenors Quantitative Arguments Are Based on a Misapplication of the Information in the BLEU Project ISAs

Intervenors misinterpret accident likelihood information presented in the ISA Summaries to be quantitative probabilities and, as a result, greatly overstate the potential frequency of accidents analyzed in the BLEU Project ISAs.

1. Purpose and Scope of ISAs

In 2000, the NRC amended the provisions in 10 C.F.R. Part 70 for the licensing and regulation of SNM facilities and those authorized to possess a critical mass of SNM.

65 Fed. Reg. 56,211 (Sept. 18, 2000). The NRC's purpose in amending the regulations was to increase confidence in the margin of safety at facilities affected by the new rule. Id. The increased confidence is to be accomplished, *inter alia*, by identifying potential accidents at each facility, defining the items necessary to prevent those accidents or mitigate their consequences, and requiring that measures be implemented to ensure that the items relied on for safety ("IROFS") are available and reliable to perform their functions when needed. Id. Thus, the emphasis of the new regulations is on enhancing the margin of safety provided by the design and operational features of the facilities.

The Part 70 amendments require license applicants to conduct ISAs to demonstrate the accomplishment of the above-described objectives. 10 C.F.R. §§ 70.61(a), 70.62(c). As its name suggests, the main purpose of an ISA is to show that safety controls are in place to reduce the likelihood of accidents to an acceptable level or to mitigate those accidents' consequences to a level below the regulations' defined thresholds. In the end, the intent is to reduce accident risk, i.e., the product of accident likelihood and consequences. Thus, accidents with potentially high consequences must be shown to be highly unlikely or to have their consequences mitigated to below the regulation's thresholds. 10 C.F.R. § 70.61(b). Also, accidents with potentially intermediate consequences must be shown to be unlikely or to have their consequences mitigated to below the regulation's thresholds. 10 C.F.R. § 70.61(c).

The ISA process, successfully completed, shows that the licensee or license applicant has comprehensively considered facility hazards that could potentially result in adverse consequences. Through the ISA process, the licensee (1) describes equipment, structures, and process activities at the facility; (2) identifies and analyzes the potential hazards at the facility; (3) identifies potential accident sequences that would result in unacceptable consequences and assesses the expected likelihood of those consequences; (4) identifies and describes the controls or safety systems (IROFS) put in place and relied on to prevent potential accidents or mitigate their consequences; and (5) identifies measures taken to ensure the availability and reliability of the identified safety systems.[18] The applicant then submits an ISA Summary to the NRC as part of its license application or amendment. 10 C.F.R. § 70.65.

Briefly summarized, the preparation of an ISA Summary proceeds as follows. For each accident sequence analyzed, the applicant assigns an Initiating Event Frequency Index value to the initiating and enabling events,[19] and also assigns Effectiveness of Protection Index values to each IROFS implemented to either prevent the accident or mitigate its consequences.[20] Two algebraic summations of the index values occur. The first summation adds only the Initiating Event Frequency Index values, to arrive at an "Uncontrolled Likelihood T Index" value. The second summation adds both the Initiating Event Frequency Index Values and the Effectiveness of Protection Index values, to arrive at the "Controlled Likelihood Index T" value. The difference between the Controlled and Uncontrolled Likelihood Index T values evidences the increased level of safety provided by the IROFS in the controlled sequence. Id. at 8. A sufficiently low Controlled Likelihood Index T value (-4 for high consequence events and -3 for

[18] NUREG-1513, Integrated Safety Analysis Guidance Document (May 2001) ("NUREG-1513") at § 2.1.

[19] The "initiating event" is the event that must occur to begin an accident sequence, e.g., valve failure allowing uncontrolled addition of HEU to a tank. Enabling events are subsequent events that must also occur for the accident sequence to proceed to the point at which adverse consequences occur, e.g., failure of the tank overflow line to vent the tank and prevent a pressure buildup. Frost Decl. at 6-7.

[20] Declaration of Jennifer K. Wheeler and Carol L. Mason Regarding NFS Response to Chemical Accident Sequences Cited by Intervenors in Their Written Presentation (Dec. 15, 2004) ("Wheeler/Mason Decl."), enclosed as Attachment 1 hereto, at 6.

intermediate consequence events) demonstrates that the licensee has implemented controls that reduce the accident sequence's risk to an acceptable level, i.e., below the specified regulatory threshold. See id. at 9. The ISA process serves to substantially improve the level of knowledge of potential accidents and the understanding of how IROFS are implemented to prevent accidents or mitigate their consequences.[21]

The ISAs and, more generally, the new Part 70 requirements are examples of risk-informed regulation.[22] According to the NRC's definition, a risk informed approach augments traditional regulation by (1) considering a broader range of safety challenges; (2) prioritizing those challenges based on risk significance, operating experience, and/or engineering judgment; (3) considering a broader range of countermeasures against the safety challenges; (4) identifying and quantifying uncertainties in the analyses performed; and (5) testing the sensitivity of the results to key assumptions.[23] However, in implementing its risk-informed approach, the NRC expressly rejected requiring licensees to use quantitative risk analyses, such as PRAs,[24] in performing the ISA. SECY-00-0111 at Attachment 9, pp. 11-12, 27.

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[21] SECY-00-0111, Rulemaking Issue Affirmation, Final Rule to Amend 10 C.F.R. Part 70, Domestic Licensing of Special Nuclear Material, Attachment 9 – Regulatory Analysis at 13 (May 19, 2000).

[22] The NRC considers the implementation of its September 2000 Part 70 amendments to be a "significant risk-informing accomplishment" because the new requirements are both risk-informed and performance based in that they require licensees to perform an ISA that identifies significant potential accidents and the implemented IROFS, and then to implement measures to ensure the IROFS are available and reliable when needed. See SECY-04-0197, Update of the Risk-Informed Regulation Implementation Plan (Oct. 25, 2004) ("RIRIP Update") at Part 1-1, Part 2, Chap. 2-28.

[23] Id. at Part 1-1.

[24] A PRA examines how engineered systems and human actions work together to ensure plant safety through a quantitative analysis that calculates the probabilities of potential events with health consequences, along with the magnitude of these consequences. See NRC, Fact Sheet, Probabilistic Risk Assessment, available at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/probabilistic-risk-asses.html (last visited Nov. 29, 2004) ("PRA Fact Sheet").

2. Intervenors' misconstrue the Controlled Likelihood Index T Values Contained in the ISA Summaries

Intervenors argue that despite the BLEU Project's engineered safety features, the potential for accidents has not been reduced to an insignificant level. Intervenors' Presentation at 2-3. In support of their argument, Intervenors assert that the Controlled Likelihood Index T values in the ISA Summaries (e.g., -3, -4, -5, etc.) represent exact quantitative probabilities corresponding to accident frequencies of 10^{-3} or 10^{-4} or 10^{-5}, etc., per accident per year. See Intervenors' Presentation at 29-31. This is a crucial error that invalidates Intervenors' claims.

The BLEU Project ISA Summaries do not estimate the probabilities of occurrence for accident sequences at the facility. Rather, as contemplated by NRC regulations, the BLEU Project's ISA Summaries provide qualitative envelopes or bounding maxima that demonstrate that potential accident sequence likelihoods have been reduced to an acceptably low level – below the defined thresholds in NRC's safety regulations. Indeed, each NRC SER prepared for each license amendment authorization describes in detail the qualitative nature of the Controlled Likelihood Index T values presented in the ISA Summaries. See UNB Amendment SER (July 2003) at 44-48; BLEU Preparation Facility SER (Jan. 2004) at 7.0-7–10; OCB/EPB SER (July 2004) at 27-31. For example, the UNB Amendment SER provides:

> NFS also applied qualitative criteria for its use of the terms, "highly unlikely" and "unlikely." Similar to NFS's application of qualitative criteria for "credible," they defined the likelihood of "highly unlikely" to be an index of -4 and "unlikely" an index of -3, instead of a frequency per accident per year. These initiating event frequencies were based on past experience, engineering judgment, analytical data, industry accepted values, and other information, if available.

UNB Amendment SER at 47 (emphasis added). Although Intervenors cite to the SERs several times, Intervenors' Presentation at 20, 26, 34, 36, they disregard the SERs' statements as to the qualitative nature of the information contained in the ISA Summaries in fashioning their arguments. While the analyses performed to arrive at the Likelihood Index T values cited by the Intervenors are risk-informed, they are do not provide quantitative probabilities for the various accident sequences.

Moreover, as discussed in detail below, the ISA Summaries' Controlled Likelihood Index T values only indicate the likelihood of accident sequences so far as necessary to satisfy NRC safety regulations under 10 C.F.R. Part 70. If an accident sequence is <u>less</u> likely than is required to satisfy the safety regulations, there is no need for the ISA Summaries to indicate that, and they do not. In other words, once the ISA Summary demonstrates that the facility meets the performance requirements of 10 C.F.R. § 70.61, the safety analysis need not and does not go any further. Indeed in most, if not all, cases analyzed in the ISAs, the accident sequence risk indices do not consider additional unlikely events that would have to occur or safety systems that would have to fail. Consideration of those additional events and safety systems would show the accident sequences to be far less likely than the indices alone suggest.

Therefore, the ISA Summaries likelihood indices do not represent the frequency of occurrence for potential accidents. They are bounding values: a Controlled Likelihood Index T value of –4 shows an approximate accident frequency of <u>less than</u> 10^{-4}, and a T value of –5 shows an approximate accident frequency of <u>less than</u> 10^{-5}, etc. Considering the conservatisms involved in the ISA analyses, but not reflected in the ISA Summaries, the actual frequencies of occurrence of the accidents, if computed, would be <u>much less</u>

than what those values would suggest. See discussion in Sections III.B & C, infra. Thus, Intervenors have greatly overstated the potential accident frequency that an individual Likelihood Index T value represents and have erroneously concluded that an EIS should be prepared because of the accident "probabilities" Intervenors infer from the ISA Summary indices.

3. Information Presented in the ISA Summaries

The first step in performing an ISA is conducting a Process Hazards Analysis ("PHA"). Wheeler/Mason Decl. at 1. The PHA identifies credible accident sequences that result from a single upset event and the controls needed to prevent them, limit their occurrence, or mitigate their consequences. Id. Next, a consequence analysis is performed to determine if the consequences for each identified accident might exceed the intermediate or high exposure levels identified in 10 C.F.R. 70.61 (b) and (c) without taking credit for any safety controls. Id. at 2. Then, a risk assessment is performed to determine the likelihood of each accident sequence identified as having intermediate or high consequences. Id. The risk assessment is used to demonstrate compliance with the performance requirements of 10 C.F.R. § 70.61. Id. Safety measures (IROFS) are added to accident sequences so that all high consequence accidents are highly unlikely, and all intermediate consequence accidents are unlikely.

However, the ISA Summaries do not summarize the entire universe of safety controls, and thus do not represent the total margin of safety in place for each accident sequence. The purpose of an ISA Summary is not to demonstrate the level to which risk has been reduced. Rather, its purpose is to demonstrate that the performance requirements of 10 C.F.R. § 70.61 have been met. Once the licensee demonstrates in the

ISA Summary that sufficient safety controls are in place for both high and intermediate consequence events to meet the § 70.61 performance requirements, the ISA is complete. There is no requirement to demonstrate, through the ISA Summary, that any additional margin of safety exists.[25]

In fact, additional safety margin is in place that is not reflected in the BLEU Project ISA Summaries. For example, the chemical consequence analyses assumed tanks and process equipment were filled to their maximum capacity, even though this would never occur. Wheeler/Mason Decl. at 14. The nuclear criticality accident analyses assumed that hand-held containers contain 12 kg of HEU when in fact they contain an average of 9 kg and no more than 11kg. Frost Decl. at 4-5. The overly conservative analyses therefore provide an even greater margin of protection. For additional discussion of the conservatisms incorporated into the safety analyses, see infra, Sections III.B. & C.

The ISA Summary presents in table format the results of the analysis that was undertaken for each credible accident scenario, to the extent sufficient to demonstrate compliance with § 70.61. See e.g., Feb. 6, 2004 BPF ISA Summary at 4-106 – 107. Intervenors have not challenged any of the analyses performed or conclusions reached in the BLEU Project ISA process, as presented in the ISA Summaries. A description of the contents of the ISA Summary table using Accident Sequence 4.1.29 as an example follows. Id.

[25] Declaration of Robert L. Frost Regarding NFS Response to Criticality Accident Sequences Cited by Intervenors in Their Written Presentation (Dec. 14, 2004) ("Frost Decl."), enclosed at Attachment 2 hereto, at 2.

Accident Sequence	Initiating Events/ Enabling Events	IROFS Effectiveness of Protection Index	Likelihood Index T Uncontrolled/ Controlled	Likelihood Category	Conseq. Category	Risk Index
4.1.29 Pump Seal Fails	Pump Seal Fails IE=0 Loss of supply pressure EE=-1	BPF-21(A) (-2) BUM-4(A) (-2)	Unc T = -1 Con T = -5	Unc = 3 Con = 1	3 3	9 3

The first column of the ISA Summary table lists each accident scenario being analyzed. In the table above, that is "pump seal fails."

The second column of the ISA Summary table lists the accident scenario initiating and/or enabling events and assigns an Initiating Event Failure Frequency Index for each of them. Id. at § 5.2.3. The Initiating Event Failure Frequency Index corresponds to the likelihood of occurrence for the initiating and enabling events[26] and is based on past experience, engineering judgment, analytical data, industry acceptable values, and any other available information. Id. at § 5.2.3.

The third column in the ISA Summary table identifies the IROFS implemented as safety controls. See e.g., Feb. 6, 2004 BPF ISA Summary at 4-106. IROFS fall under one of several categories, depending on the nature of the safety function performed, e.g., passive engineered, active engineered, enhanced administrative, or administrative. See Wheeler/Mason Decl. at 6. Each IROFS is assigned an Effectiveness of Protection Index value, which qualitatively indicates the level of protection provided by the IROFS. Feb. 6, 2004 BPF ISA Summary at 5-8. The Effectiveness of Protection Index value

[26] In some cases the Initiating Event Failure Frequency Index represents the collective likelihood of the initiating and enabling events required to cause a particular accident sequence to occur. See, e.g., OCB/EPB ISA Summary Rev. 0 (October 2003) at 225, "External water into blending system – Water used to fight a large fire inside the ModCon area." For some accident sequences, there is no Initiating Event Failure Frequency Index, indicating that the initiating event(s) are expected to occur approximately once a year or regularly during the facility's lifetime. See, e.g., ISA Summary Rev. 3 (January 2004) at 26, Accident Sequence 1.5.2.

assignment is based on industry-accepted values, past operating experience, engineering analysis, analytical data, and any other applicable information. Wheeler/Mason Decl. at 5.

In example 4.1.29 in the above table, both IROFS were assigned a "-2" effectiveness of Protection Index value. A "-2" value indicates that the particular IROFS employed provides the level of protection afforded by a single functionally tested active engineered control or a trained operator performing a routine task with an approved procedure, an enhanced administrative control, or an administrative control with a large margin of safety, in conjunction with adequate management measures to ensure its availability. Feb. 6, 2004 BPF ISA Summary at Table 5-4. A "-3" Effectiveness of Protection Index value provides the level of protection afforded by an inspected single passive engineered control or exceptionally robust functionally tested active engineered control with a trained operator backup and adequate management measures in place to ensure its availability. Id.

NFS modeled its "IROFS Effectiveness of Protection Indices" (shown in column three in the ISA Summary table) after those contained in Table A-10 of NUREG-1520,[27] the NRC's Standard Review Plan for fuel cycle facilities. Wheeler/Mason Decl. at 7. A "-2" Effectiveness of Protection Index value corresponds to a "-2 or -3" index value in Table A-10. The -2 to -3 range in Table A-10 correlates to a probability of failure on demand of $10^{-2} - 10^{-3}$. Id. at 8; compare NUREG-1520 at 3-A-11. When NFS was developing its ISA Summary presentation, it used the most conservative (i.e., least negative) index value. Wheeler/Mason Decl. at 8. Thus, for example, the NFS IROFS

[27] NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, Final Report (Mar. 2002).

Effectiveness of Protection Index value of -2 envelops a probability of failure on demand between $10^{-2} - 10^{-3}$.

Column four of the ISA Summary table provides the Controlled and Uncontrolled Likelihood Index T values (described in the previous section). The Controlled Likelihood Index T value is what the Intervenors mistakenly interpret as reflecting the probability per year that an accident sequence will occur. See Intervenors' Presentation at 29 n.17.[28]

The Controlled Likelihood Index T values for an accident sequence are given a descriptive Likelihood Category designation in the fifth column of the ISA Summary table. See Feb. 6, 2004 BPF ISA Summary at Tables 5-6, 5-7. For example, a Controlled Likelihood Index T value less than or equal to -4 is assigned a Likelihood Category of 1, or "Highly Unlikely." Id.[29] In the above table for Accident Sequence 4.1.29, the Controlled Likelihood Category is Con=1, or "highly unlikely."

The ISA Summary's sixth column contains the Consequence Category, which reflects the conservatively estimated severity of the unmitigated consequences of the accident, should it occur. Feb. 6, 2004 BPF ISA Summary at 4-107. Table 5.7, Risk Matrix, provides definitions for the Consequence Categories; they range from category 1 (low) to 2 (intermediate) to 3 (high). Id. at 5-11. By multiplying the Likelihood

[28] As discussed below, all of the accident sequences cited by Intervenors require additional unlikely events to occur and/or additional safety systems to fail before the sequences would occur. See, e.g., Frost Decl. at 4-5; Wheeler/Mason Decl. at 13-14. Thus, the actual likelihoods of the accident sequences are significantly lower than what is indicated by the Likelihood T Indices alone, and the probabilities of the accident sequences are significantly lower than what the Intervenors assert them to be. Id.

[29] The conservatism in NFS' Effectiveness of Protection Index numbers explains why it defines a –4 index value as "highly unlikely." NRC's guidance defines highly unlikely as a –5 Likelihood Index T value. NUREG-1520 at 3-A-9. However, NUREG-1520 also permits the use of less conservative (more negative) Effectiveness of Protection Index numbers to demonstrate the same envelope of protection. Id. at 3-A-11. Thus, summing more negative index value numbers would result in more negative Likelihood Index T values, even though the same safety envelope has been established. Wheeler/Mason Decl. at 5-6.

Category (controlled and uncontrolled) by the Consequence Category, the controlled Risk Index is determined, as listed in column seven of the ISA Summary table. Id. at 4-107. This product determines whether the controlled accident risk is low enough to be accepted under 10 C.F.R. § 70.61.

As shown in the table above, accident sequence 4.1.29 has a Controlled Likelihood Category of 1 (highly unlikely) and Consequence Category of 3 (high consequences), the product of which equals a controlled Risk Index of 3. Risk Index values of 1 through 4 indicate risk of an "acceptable" level. Feb. 6, 2004 BPF ISA Summary at 4-107. The difference in value between the controlled and uncontrolled indices again reflects the level of reduction in risk provided by the IROFS. For example, in accident sequence 4.1.29, the accident sequence has the potential for high consequences, but the overall risk is reduced to an acceptably low level in the controlled sequence because the two IROFS have been implemented and have significantly reduced the sequence's likelihood of occurrence.

B. Intervenors Greatly Overstate the Likelihood of Potential Accidents

Intervenors claim that "NFS's own ISA Summaries demonstrate that severe criticality accidents, chemical spills, fires, and other accidents are reasonably foreseeable under the NRC's own standards because NFS's quantitative definition of what constitutes an 'unlikely' or 'highly unlikely' accident falls within the range of accident probabilities considered to be reasonably foreseeable by the NRC." Intervenors' Presentation at 28. They claim that NFS has defined "unlikely" as "[n]ot expected to occur during the plant lifetime" and that NFS has assigned a "quantitative probability" of "less than or equal to

10^{-3} per accident per year." Id. at 29 (citing October 11, 2002 ISA Summary for aluminum dissolution and downblending process in the BPF at 9-1). Intervenors claim that NFS has defined "highly unlikely" as "[p]hysically possible or credible, but not expected to occur" and has assigned to that "a quantitative value of less than or equal to 10^{-4} per accident per year." Id. Intervenors then go on to claim that accidents with those probabilities give rise to environmentally significant risk. See id. at 31-32.

As shown below, Intervenors have grossly overestimated the probabilities of occurrence of accidents associated with the BLEU Project. Intervenors' have misunderstood the ISA process and have thus misinterpreted the accident likelihoods presented in NFS's ISA Summaries. By significantly overestimating accident probability, Intervenors have overestimated accident risk and thus the environmental significance of the potential accidents associated with the BLEU Project.

1. Criticality Accidents

Intervenors assert that NFS's ISA Summaries "provide estimates as high as 10^{-4} and 10^{-5} per accident per year" for "some high-consequence criticality accidents." Id. Intervenors' claim is based on the "Controlled Likelihood Index T" values that NFS's ISAs assessed for the BLEU Project's potential criticality accident sequences (asserting that -4 and -5 correspond to probabilities of 10^{-4} and 10^{-5}, respectively). See id. at 29 & n.17. Intervenors cite several criticality accident sequences for which Controlled Likelihood Indices of -4 and -5 were assessed. See id. at 29-31 (citing individual accident sequences). They assert that because of their probabilities, the risk posed by those accidents gives rise to significant environmental impacts. Id. at 31-32.

In discussing criticality accidents, like elsewhere in their presentation, Intervenors misconstrue the ISA process and thus overestimate the probabilities of criticality accidents associated with the BLEU Project. Intervenors' assertion that the criticality accident sequences that they cited have "probabilities" of 10^{-4} or 10^{-5} per year is erroneous because it fails to take into account that (1) the ISA likelihood indices are conservative upper bound estimates, not estimates of probability and (2) once the ISAs demonstrate that the criticality accident sequences are highly unlikely, the analysis stops and does not go on to assess the actual (lower) probability of each sequence. Frost Decl. at 1.

The discussions below and in Dr. Frost's Declaration demonstrate the conservatism of the ISA evaluations and point out all the events that would have to occur beyond those credited in the ISA before a criticality accident would be possible. Following is a detailed discussion of two criticality accidents to illustrate the conservatism of the ISA assessments and the fact that the accident probabilities are significantly lower than what the Controlled Likelihood indices in the ISA Summaries alone would suggest. These principles hold true for all of the criticality accident sequences cited by the Intervenors.[30]

a) HEU Container Spacing Violations

Intervenors cite (Intervenors' Presentation at 29-30) three criticality accident sequences involving the mishandling of HEU containers in the BPF Uranium Metal Dissolution Process Area:

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[30] All of the criticality accident sequences cited by the Intervenors are addressed in Dr. Frost's Declaration; similar sequences are grouped together to facilitate discussion. See Frost Decl. at 2.

- 4.1.26.4.1 Container spacing upset with process equipment with only one operator handling portable containers (ISA likelihood index of -4)
- 4.1.26.4.1.b Container spacing upset with process equipment with only one operator handling portable containers (ISA likelihood index of -4)
- 4.1.26.4.2 Container spacing upset with storage racks with two or more operators handling portable containers (ISA likelihood index of -4).

Frost Decl. at 3. These accident sequences are examples of sequences prevented to a significant part through the use of administrative controls. Examination of how the ISA process assessed the likelihoods of such accidents shows how conservative the assessments are.

The use of small bottles and cans to store and transport HEU is a common feature of all facilities that process HEU. Id. Safe procedures for storage and transport of these containers have been established and are observed throughout all NFS operations. Id. These procedures are designed to assure a minimum separation distance is maintained between containers and between containers and other equipment that may contain HEU. Id.[31]

All three accident sequences involve violations of the controls (IROFS) that assure containers remain properly spaced. Id. Those controls are:

1. Only one container may be hand-carried per person at a time.
2. Hand-carried containers must be spaced at least 12 inches from each other and from process equipment.

Criticality is theoretically possible if an operator were to hand-carry two or more containers simultaneously, in violation of requirement number 1, and then place his hand-carried containers in contact with each other and with a piece of equipment containing

[31] Spacing prevents the assembly of a critical mass and geometry of HEU.

HEU. Id. By contrast, if the operator were to hand carry only a single container and, in violation of requirement number 2, placed that container in contact with one other container or with equipment containing HEU, no criticality would occur. Id.

In order for this accident sequence to occur, the operator must therefore commit multiple violations of criticality safety requirements:

1. He first must pick up and carry more than one container, in violation of requirement (1) above.
2. He must ignore the 12-inch spacing requirement for those containers, in violation of requirement (2) above.
3. Finally, he must ignore the 12-inch spacing requirement between containers and process equipment, again in violation of requirement (2) above.

Each of the three steps listed above is a violation of an IROFS. Id. at 4. These administrative controls are routinely applied by operators in their daily operations, and are backed up by extensive operator training. Id. This makes assignment of a -2 Effectiveness of Protection Index for each of these IROFS appropriate. Id. NFS assigns a -2 to the index representing an IROFS administrative control, "protected by a trained operator performing a routine task with an approved procedure." This correlates to the information presented in Table A-10 of NUREG-1520 (at 3-A-11), where an index of -2 is roughly equivalent to a probability of failure on demand of 10^{-2}. Wheeler/Mason Decl. at 7. This value is supported as a conservative assessment by the Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities;[32] see Wheeler/Mason Decl. at 8-9.[33] Summing the effectiveness of protection indices leads to

[32] Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities (DE94012947), dated February 28, 1994.

[33] Failure rate data show that the mean probability of failure of the administrative controls is less than 10^{-2} and that probability is further reduced by NFS' application of Management Measures to the controls to

a -4 Controlled Likelihood Index for the accident sequence. Frost Decl. at 4. The -4 index corresponds to a determination that the accident sequence is highly unlikely and meets the requirements of 10 C.F.R. § 70.61. Id.

However, the actual accident probability is at least an order of magnitude lower than the "highly unlikely" determination from the ISA because even violations of both IROFS would not necessarily cause criticality. Id. at 4-5. First, safety control number 2 is that a 12-inch separation must be maintained between containers of HEU, or between such containers and process equipment that contains HEU. Id. at 4. Violation of this requirement is assumed to result in no separation between the containers and equipment. Id. This is a very conservative assumption because, in fact, the separation must be less than half an inch for criticality to be possible. Id. It is much less likely that the containers and equipment would be accidentally placed within half an inch of each other than within 12 inches of each other. Id. Nevertheless, in the ISA process a -2 Effectiveness of Protection Index was conservatively assigned to the failure to maintain the required 12 inch spacing; there was no distinction between a small, inconsequential violation (e.g., 11 inch spacing) and a significant one (1/2 inch or less spacing). Id.

The ISA made the additional conservative assumption that the containers of HEU and the equipment all contained 12 kg U of HEU, when in fact this is never the case. Id. at 4-5. The maximum capacity of a container is 11 kg U of HEU (and fewer than 1% of them actually hold that much) and on average they contain only 9 kg U. Id. at 5. Thus, it is extremely unlikely that the two containers involved in an accident would both contain 11 kg U. Id. Further, with an average container containing 9 kg U, an operator would

ensure that they are effective. Wheeler/Mason Decl. at 9.

have to carry three containers, hold them together, and place them in contact with HEU-bearing equipment, for criticality to occur. Id. At 9 kg U (~20 pounds) per can, carrying three cans, weighing 60 pounds, would present a considerable physical impediment to such a maneuver. Id. Furthermore, for criticality to occur, the equipment with which the containers are brought into contact must also contain a significant amount of uranium. Id. Much of the equipment in the BPF operates in batch mode, and therefore is sometimes empty or in the process of being loaded, with only a small amount of uranium present. Id. All of these factors show that a criticality accident would be unlikely even if two containers and HEU-containing equipment were brought into contact. Id.

In conclusion, the ISA analysis assumes that only three violations have to occur for a criticality accident to result. In fact, two further unlikely events not credited in the analysis must also occur:

4. The containers must be brought in close contact such that all are within ½ inch of each other.

5. Both of the containers must contain 11 kg of material (or an operator must carry three HEU containers at once) and at the same time the HEU-containing equipment must contain a significant amount of material.

Id. Thus, the likelihood of this scenario is much lower than even the -4 Controlled Likelihood Index assessed by the ISA suggests, and the actual probability of occurrence of this accident sequence is at least an order of magnitude lower than the 10^{-4} asserted by the Intervenors.

b) Backflow of Fissile Solution Into Plant Air System

Intervenors cite (Intervenors' Presentation at 30) two criticality accident sequences involving pump seal failure in the U Metal Dissolution Process Area:

- 4.1.28 Pump seal fails (ISA likelihood index of -5)
- 4.1.29 Pump seal fails (ISA likelihood index of -5).

Frost Decl. at 6. (The accident sequence Backflow-2 (ISA likelihood index of -5), in the OCB Uranium Recovery Process Area (cited in Intervenors' Presentation at 30), is very similar to the two pump seal failure accident sequences discussed here.) Id.

These accident sequences refer to HEU backflow into the Plant Air supply system. Id. In such an accident, HEU would flow from a favorable geometry column, in which HEU solution is processed or stored, back through a Plant Air supply line into the Plant Air supply system. Id. It should be noted that backflow into other utilities or into chemical supply systems would require the occurrence of very similar events and control failures. Id. Thus, this discussion is also applicable to potential accidents involving HEU backflow into those systems as well. Id. These are examples of accident sequences prevented largely or entirely through the use of engineered controls (as opposed to administrative controls), so this discussion is illustrative of how the ISA conservatively assessed the likelihoods of accidents managed through engineered controls.

Some of the operations in the BLEU Project facilities utilize "favorable geometry columns" to store or process HEU solution. Id. A favorable geometry column has a small diameter, such that criticality is not possible, regardless of the column height or the concentration of uranium in the solution. Id. This is a very robust passive engineered control. Id. In some cases these columns are serviced by utilities, such as plant air, or by chemical supply lines. Id. The utility and chemical supply lines are also very small in diameter and therefore of favorable geometry, but often lead to large tanks that are not of favorable geometry. Id. Therefore, it is necessary to provide means to assure HEU solution will not backflow from the favorable geometry columns into the utility or chemical supply lines. Id.

In order for uranium-bearing solution to backflow into the Plant Air system, the solution pressure must exceed the pressure in the Plant Air system, and any barriers to backflow must be removed.[34] The accident sequence can be described as follows (see Frost Decl. at 6-7 for more detail).

1. Uncontrolled addition of HEU solution to favorable geometry column. This requires valve failure or operator error in leaving it open. Such an event is expected to occur only a few times during the life of the facility.

2. Failure of the overflow line to vent the favorable geometry column, thereby allowing the column to become pressurized. The overflow line is a robust passive engineered control. Despite the highly robust nature of the overflow line as a means of preventing backflow, and the resulting low likelihood of this event, it is conservatively not credited in the ISA.[35]

3. Failure of the Plant Air supply system, such that the pressure in the Plant Air lines reduces to near atmospheric levels. Variations in Plant Air pressure are expected, but failure of the system to very low levels is not a common occurrence, expected to occur a few times during the life of the facility.[36] However, this low frequency is also conservatively not credited in the ISA.

4. Failure of the diaphragm on the drain line transfer pump, which removes the barrier between the process solution and the plant air line. Such a failure is expected to occur with a low frequency, due to periodic maintenance on the pump and the requirement that the diaphragm material of construction be compatible with the chemicals being pumped. The combination of the initiating event (step 1) and this enabling event is assigned a conservatively high frequency index of -1.

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[34] Frost Decl., Figure 1 illustrates a simplified arrangement of a favorable geometry column that is supplied uranium-bearing solution from another favorable geometry source. The column contents are pumped out through the drain line using an air diaphragm pump that is supplied by the Plant Air system. An overflow line on the column is vented to atmosphere, thereby assuring that the column contents are normally at atmospheric pressure.

[35] Passive engineered controls, if relied upon as IROFS, are typically assigned Effectiveness of Protection Indices of –3, which correspond to probabilities of failure on demand of 10^{-3} to 10^{-4}. See Wheeler/Mason Decl. at 6; NUREG-1520 at 3-A-11 (Table A-10).

[36] As with step 1, this event would have a failure frequency index of –1 or a probability of approximately 10^{-1} per year.

5. The first IROFS is a pressure sensor interlocked to a pneumatic valve. The pressure sensor has a set point of 70 psi. If the Plant Air supply pressure drops below 70 psi, the valve automatically closes. It must fail for the accident to occur. This active engineered feature is conservatively assigned a failure index of -2.[37]

6. The second IROFS is a second, independent pressure sensor/interlocked valve. It must also fail for the accident to occur. It is also conservatively assigned a failure index of -2.

The ISA determined that this accident sequence had a Controlled Likelihood Index of -5, based on the fact that both IROFS, with Effectiveness of Protection Indices of -2, had to fail for the sequence to occur and the initiating/enabling event index is -1. Frost Decl. at 8.[38] However, the likelihoods of the other two enabling events are not credited in the assessment. Id. The probability that the initiating event and three enabling events (events 2-4 above) would all occur concurrently is so low that the accident sequence probably is not even credible. Id. For example, if an index of -3 was assigned to the passive engineered control and the remaining uncredited enabling event, then the total controlled likelihood index would be -8 or less. That would correspond to a probability of roughly 10^{-8} per year or less, which is well below the point of credibility.

There are additional conservatisms in the ISA's likelihood assessment. Id. Backflow of uranium solution into the Plant Air system, by itself, does not assure that criticality will occur. Id. Criticality would only be possible if the accumulation of the uranium-bearing solution in the Plant Air tanks became significant. Id. It would take a long period of time for such a buildup to occur, during which an operator would be

[37] Active and passive engineered controls were assigned Effectiveness of Protection Indices based on industry accepted values, past operating experience, engineering analysis, analytical data, and/or other applicable information (e.g., industry standards such as those produced by the Institute of Electrical and Electronics Engineers). Wheeler/Mason Decl. at 6-7; see also NUREG-1520 at 3-A-11 (Table A-10).

[38] The Intervenors, reading the Controlled Likelihood Index value off the ISA Summary Table, describe the sequence as having a probability of 10^{-5} per year (Intervenors' Presentation at 30).

expected to notice the problem and close a valve to terminate the backflow. Id. Such an accident avoidance measure, which renders this accident sequence even more unlikely, is also not credited in the ISA analysis. Id.

These two accident sequences illustrate how the ISA process works and why Intervenors were gravely mistaken in assuming that the likelihood indices in the ISA summaries corresponded to the orders of magnitude of the probabilities of the accident sequences the ISA assessed. For these accident sequences, the ISA assessed controlled likelihood indices based only on the two IROFS. Having determined that the Controlled Likelihood Indices total indices were –5, the assessment simply stopped because it had already satisfied the requirement of 10 C.F.R. § 70.61 to demonstrate that the sequences were highly unlikely, see Wheeler/Mason Decl. at 7 (Tables 4 and 5). It did not go on to further incorporate other events into the controlled likelihood indices, even though there were three unlikely enabling events in the sequences that could have been assessed and would have further reduced the indices (indeed, one event could have been assigned an index of -3).

Indeed, in this case the actual accident probabilities, if computed, would be shown to be several orders of magnitude lower than what the ISA indices suggest. As discussed below, that was the case with every accident sequence the ISA assessed. Therefore, the Intervenors are simply wrong in concluding from the ISA Summary Controlled Likelihood Indices that the probabilities of high consequence accidents associated with the BLEU Project are on the order of 10^{-4} or 10^{-5} per year. In fact, they are orders of magnitude lower and do not give rise to environmentally significant risk.

c) **Other Criticality Accident Sequences**

In his Declaration, Dr. Robert Frost goes on to address all of the criticality accident sequences cited by the Intervenors. Dr. Frost describes the accident sequences, discusses how the ISA assigned controlled likelihood indices to them, and demonstrates the conservatism of the ISA assessment. In every case, it was not necessary that the ISA credit applicable factors that rendered the event more unlikely and/or additional safety systems or controls, because they were not necessary to show compliance with 10 C.F.R. § 70.61.

For example, with respect to the potential addition of excess uranium to enclosures in the uranium-aluminum dissolution area, the assessment assumes that four ingots in an enclosure would result in criticality, when in reality at least six would be necessary. Frost Decl. at 11. With respect to the potential crystallization of uranium solution in the UNB receipt tank (TK-10), the assessment conservatively does not consider the likelihood that the solution would freeze during transport to NFS, or that criticality could be prevented by the condition of the solution in the tank before the potential addition of the crystallized solution. Id. at 14-15. With respect to uranium precipitation in the receipt tank, the assessment does not account for the fact that experiments have shown that uranium precipitation would not increase the uranium concentration in the solution and hence would not cause criticality. Id. at 15-17. With respect to the potential transfer of effluent with a high concentration of uranium into the EPB, the assessment assumed that the accident initiating event was the transfer of solution with a concentration of uranium of over 1 ppm, when in fact a concentration orders of magnitude higher is required for criticality to result. Id. at 24-25.

In other words, the probabilities of the criticality accident sequences cited by the Intervenors are orders of magnitude lower than the 10^{-4} or 10^{-5} per year that they assert based on the Controlled Likelihood Indices contained in the ISA Summaries. Thus, those accident sequences do not give rise to environmentally significant risk.

2. Chemical Accidents

Intervenors also assert that the ISA Summaries "list a number of chemical accidents with high consequences." Intervenors' Presentation at 31.[39] They claim that "according to the [NRC Staff] SERs" the likelihood of these accidents "has been reduced to a level that is 'highly unlikely,' i.e., 10^{-4} per accident per year." Id. Intervenors also assert that several chemical accident sequences were assessed as having "intermediate" consequences and that according to the NRC Staff SERs, the probabilities of those accidents "has been reduced to an 'unlikely' level, i.e., 10^{-3} per accident per year." Id. at 32 (citing First SER at 48, Second SER at 7.0-13, Third SER at 34). Intervenors claim that because of their probabilities, the risk posed by these accidents gives rise to significant environmental impacts. Id. at 31-32.

Contrary to Intervenors' claims, the probability of a chemical accident associated with the BLEU Project actually having high consequences is significantly less than 10^{-5} per year, and the probability of a chemical accident actually having intermediate consequences is significantly less than 10^{-3} per year. Intervenors make the same error with respect to potential chemical accidents as they do with respect to potential criticality

[39] Intervenors cite Table 4-4 of the November 14, 2003 ISA Summary for the OCB and EPB, Table 4-5 of the February 6, 2004 Revised ISA Summary for the BPF, and Table 4-5 of the October 11, 2002 ISA Summary for the uranium-aluminum dissolution and downblending processes in the BPF. Intervenors' Presentation at 31, 32.

accidents—they ignore the fact that the Controlled Likelihood Indices are not intended to establish the probability of an accident. Moreover, they ignore the conservatism of the ISA assessments and the fact that once the ISA concludes that a high consequence accident is highly unlikely or that an intermediate consequence accident is unlikely, it does not go on to further refine its Controlled Likelihood Index. As shown below, a realistic estimate of the probabilities of high consequence chemical accidents associated with the BLEU Project would be much lower than the values of the ISA Controlled Likelihood Indices alone. The majority of the chemical scenarios identified in the Intervenor's Presentation fall into two categories discussed in subsections (a) and (b) below. The remaining scenarios are process-specific and are discussed as a group in subsection (c).

a) Chemical Leaks and Spills from Tanks and/or Piping

(1) Ammonium Hydroxide Leaks or Spills

One group of chemical accident scenarios cited by Intervenors involve the possibility of exposure to ammonia fumes due to a leak or rupture of the Ammonium Hydroxide supply header, the Bulk Ammonium Hydroxide tank, or associated supply piping for the OCB/EPB, which would result in a spill of ammonium hydroxide.[40] Leaks could also occur due to an excessive off-loading rate or fill rate from the recycle line when filling the bulk tank. Wheeler/Mason Decl. at 11.

The use of piping and tanks to store and transport chemicals is a very common feature of chemical processing facilities. Id. Design codes and practices have been

[40] Evaluations 3, 32, and 49 (from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03). "Evaluations" are bounding accident sequences – accident scenarios of a similar type where all failure modes result in consequences within the same consequence category (all high, all intermediate, or all low). These evaluations concern accident sequences involving potential ammonium hydroxide leaks or spills.

established by nationally recognized professional organizations and are utilized for NFS designs, including those associated with the BLEU Project. Id. These codes and practices have been developed to assure safe design, therefore reducing risk to the public. Id.

In order for these accident sequences to actually occur, some combination of the following events would be needed:

1. The design would fail to consider design codes and practices to include selection of the wrong materials for the supply piping and/or tanks.
2. The installation of the supply piping would fail to follow standard installation methods.
3. Hydrotesting on the supply piping and/or tanks would not be performed properly or at all.
4. The equipment would not be properly maintained after installation.
5. Operators would have to fail to operate equipment, respond to alarms, and follow operating procedures properly when filling the bulk tank.

Id. at 11-12.

Because some or all of these events would have to occur to make the equipment vulnerable to leaks, spills, or ruptures, an Initiating Event Failure Frequency Index of at least -1 was assigned to the individual accident sequences included in the pertinent evaluations. Id. at 12. These accident sequences (unmitigated) were evaluated as having high consequences. Id. Thus, an initiating event failure frequency index of -1 was not sufficient, by itself, to satisfy the requirements of 10 C.F.R. § 70.61. Therefore, IROFS were assigned to the accident sequences to make them highly unlikely. Id. The IROFS assigned to the systems involved in the ammonium spill accident sequences include the following:

- Correct installation of piping and tanks (includes material selection, fabrication methods, and hydrotesting) to prevent pipe or tank failure due to corrosion and/or structural failures

- Maintenance program to ensure that equipment is properly maintained and prevents exposure to chemical liquids or fumes due to pipe or vessel corrosion or failure

- Operating procedures and training to ensure that ammonium hydroxide pump recirculation line valve remains open during pump operation to prevent potential equipment damage

- Operator response to tank high level alarm to prevent overflows

- Operating procedures and training to prevent operator from allowing vendor to off-load to tank unless adequate volume is available for product.

Id. These IROFS are classified as Passive Engineered Controls, Enhanced Administrative Controls, or Administrative Controls. Id. Therefore, a -2 IROFS Effectiveness of Protection Index was assigned to the majority of the IROFS. Id. As noted previously, Effectiveness of Protection Indices were assigned to IROFS based on industry accepted values, past operating experience, engineering analysis, analytical data, and/or other applicable information. The Initiating Event Failure Frequency Index for the accident sequences was at least –1 and the assignment of at least two IROFS with Effectiveness of Protection Indices totaling at least -4 to each sequence resulted in at least a -5 Controlled Likelihood Index for these scenarios. Id. at 12-13. Thus, with the IROFS in place, these accident sequences were deemed to be highly unlikely and the 10 C.F.R. § 70.61 performance requirements were met. Id. at 13. The ISA assessment alone would thus suggest that these accidents have probabilities of less than 10^{-5} per year.[41]

[41] Intervenors' state that the "highly unlikely" assessment cited in the NRC Staff SERs corresponded to a 10^{-4} accident probability. Intervenors' Presentation at 31. The SER statement was based on an error in a now superseded earlier NFS ISA document. As discussed in the Wheeler/Mason Declaration (at 9), "highly unlikely" is equivalent to a quantitative probability of less than or equal to 10^{-5} per accident per year. Moreover, for these specific accidents the ISA Summary controlled likelihood indices (-5 or less)

However, the risk from these potential accidents is even lower, because the occurrence of any of the listed failures does not mean that catastrophic failure of the supply piping or tank and a release of a large quantity of chemicals—and hence a high consequence event—will necessarily follow. Wheeler/Mason Decl. at 13. In fact, it is very unlikely that a chemical spill large enough to affect the public or environment would ever occur. Id.

First, the consequence evaluations were based on the conservative assumption that a leak/spill would occur when tanks and process equipment are filled to their maximum capacities. Id. In fact, this is ordinarily not the case. Id. A leak or spill from a partially full tank would reduce the amount of chemical available to be spilled as a result of any accident. Id.

Second, several IROFS would mitigate spill consequences and thereby further reduce the probability that a spill would be a high consequence event. Id. For example, operator response to a high tank level alarm would stop an overflow before the entire tank's contents were released. Id. Maintenance programs ensure that small leaks are identified and repaired before catastrophic tank or pipe failures can occur. Id. Finally, pump recirculation lines prevent pump damage, thereby reducing the potential for catastrophic pump failure and spills of process solutions. Id.

Third, if a leak were to begin, further controls are in place to prevent it from developing into a large event. Id. These include the existence of a dike containment area for the bulk supply tank, and the activation of the NFS Spill Response Plan used for mitigating chemical spills. Id. Activation of the Spill Response Plan would contain the

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show that their probabilities would be less than 10^{-5} per year.

spill long before the public or the environment could be significantly affected. Id. These mitigating factors were conservatively not included in the likelihood assessments for these accident sequences in the ISA. Id.

Fourth, several conservative assumptions make it unlikely that large amounts of ammonia vapors would be dispersed to the environment or that large exposures to individuals would result—and thus that the consequences of the event would be high—even if a large spill were to occur. Id. All accident sequence evaluations were based on conservative assumptions: chemical evaporation rates, outdoor temperatures, wind speed and direction, and atmospheric stability for dispersion. Id. Each of these assumptions was intended to maximize the calculated amount of ammonia vapor dispersal to the public and the environment. Id. at 14.

Fifth, it was also conservatively assumed that individuals were stationed for one hour at the closest site boundary and that the wind was blowing in their direction. Id. at 4. This is extremely conservative because it is unlikely that people will be located at the site boundary and it is practically inconceivable that, even if they were, they would remain within the plume of a chemical release for one hour without leaving on their own or being instructed to leave by NFS or off-site emergency response personnel. Id. Moreover, given the large distances from the NFS site to residences, exposure to even the closest members of the public would be over 10 times less than the exposure to a hypothetical individual located at the site boundary. Id. This by itself could well reduce a "high" consequence event to "intermediate" or an "intermediate" consequence event to "low." Id.

Each of the foregoing assumptions leads to an overestimate of exposures because the intent of the ISAs was only to demonstrate that the regulatory thresholds have not been exceeded. Use of more realistic assumptions would significantly reduce the estimated consequences for the chemical accident scenarios and thus would significantly reduce the probability of a high consequence chemical accident. Id. at 13-14. Therefore, in the end, the probability of a high consequence ammonium hydroxide leak or spill would be much less than the 10^{-5} per year suggested by the ISA Summary alone.[42]

(2) Other Evaluations Involving Chemical Leaks or Spills

Intervenors cite a number of other BLEU Project accident sequences involving chemical leaks and spills that are similar in nature to the ammonia spills discussed above. Wheeler/Mason Decl. at 14. The accidents include nitric acid spills, deionized water overflow of tanks, liquid waste spills, hydrogen peroxide spills, and caustic tanker spills. Id. (citing Table 4-5 of the ISA Summary for UAL and Downblending, Rev. 0, 10/11/02, Table 4-5 of the ISA Summary for BPF, Rev. 1, 2/6/04, and Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03). All of these spills require the same or similar combination of events to occur, have the same or less frequent likelihood indices, and have the same or similar IROFS assigned to them. Thus, the ISA assessment alone would suggest that these accidents have probabilities of on the order of 10^{-5} per year. Id.

[42] Although Intervenors cite no intermediate consequence accident scenarios, the foregoing discussion also applies to such scenarios. Thus, their probabilities are well below 10^{-3} per year. Furthermore, because of the conservative assumptions made and mitigating factors not credited with respect to consequences the likelihood of an "intermediate" accident resulting in intermediate consequences (i.e., exposure for up to 1 hr without experiencing or developing irreversible or other serious health effects) to a member of the public is much less than 10^{-3} per year.

Furthermore, also similar to the ammonia accidents, several mitigating factors were conservatively not included in the risk assessment for these accident sequences/scenarios, which would further reduce the probability of a high-consequence accident. Id. These conservative assumptions include:

- Leaks/spills occurring when tanks and process equipment are filled to their maximum capacities
- Leak and spill amounts not limited by IROFS to less than maximum process equipment capacities
- Spill containment areas and Spill Response Plan not taken into account
- Conservative atmospheric dispersion assumptions (evaporation rate, temperature, wind stability and direction)
- Exposed individual located at site boundary and exposed for one hour.

Id. at 14-15. The effect of all of these conservative assumptions is that the probability of a high consequence chemical accident associated with the BLEU Project is much lower than the 10^{-5} per year suggested by the ISA Controlled Likelihood Indices alone.

b) Ammonia Vapor Release Due to Fire

The other general class of chemical accidents cited by Intervenors involve the possibility of exposure to ammonia vapors due to a fire in the OCB Tank Gallery, a fire in the EPB, or a fire on the second floor of the OCB resulting in a release of ammonia vapors from tanks or equipment. Id. at 15 (citing Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03). In order for these accident sequences to occur, some combination of the following events would be necessary.

1. Employees would have to bring combustible materials into areas where they are strictly prohibited.

2. A fire initiator would have to be in the presence of combustible material long enough to start a fire.

3. A fire would have to burn unnoticed long enough to move into the areas of concern and affect the tanks or equipment such that ammonia vapors could escape.

4. The fire suppression/detection systems would have to fail to activate thus allowing the event to continue indefinitely, potentially breaching the building and allowing the vapors to escape to the environment.

Id.

Some or all of these events would have to occur to make the equipment vulnerable to release of vapors if a fire occurred. Id. Thus, an Initiating Event Failure Frequency index of -1 was assigned to the individual accident sequences. Id. Because these accident sequences were evaluated as having high (unmitigated) consequences, IROFS were assigned to make them highly unlikely. Id. at 15-16.

The IROFS assigned and the protection they provide include the following:

- Combustible loading program restricting the amount of potentially combustible material in the operating spaces of the facilities
- Fire protection test, maintenance and inspection activities to detect and remove potential combustibles from the operating spaces of the facilities.

Id. at 16. Both of these IROFS are classified as Administrative Controls, therefore a -2 IROFS Effectiveness of Protection Index was assigned to each. Id. The Initiating Event Failure Frequency Index for the accident sequences was -1 and the assignment of at least two IROFS with Effectiveness of Protection Indices totaling -4 to each sequence resulted in a -5 Controlled Likelihood Index for these scenarios. Id. Thus, the ISA assessment alone could be misread (as Intervenors do) to suggest that these accidents have probabilities on the order of 10^{-5} per year. However, their actual probability would be well below that figure because the ISA did not credit the fire suppression/automatic sprinkler systems for the OCB/EPB and a fire detection system for the second floor of the OCB with the potential for detecting and extinguishing a fire. Id. The more likely outcome of this event would be a small fire that could be identified and stopped before it developed into a large fire with significant consequences. Id. at 16-17.

In addition, as with the other chemical spill accidents, the risk from these accidents is even lower because the occurrence of the failures listed in the ISA does not mean that the maximum release of ammonia vapors to the environment or a high consequence accident will follow. Id. at 16. In fact, it is very unlikely that, even if these accident sequences took place, a high consequence release of ammonia vapors would occur. Id.

Several factors pertaining to these accidents would further reduce the probability of a high-consequence event. Id. at 16-17. First, if the fire suppression systems functioned correctly and contained the fire (even if they did not extinguish it), the vapors would be released from the elevated stack, resulting in significant dispersion and reduced concentrations to which affected members of the public would be exposed. Id. Even if the fire remained uncontained, the additional plume rise generated by the heat of the fire would also result in reduced concentrations to which the environment and the public would be exposed. Id. at 17.

Second, the consequence assessments conservatively did not account for the mitigating effects of the NFS Emergency Response Plan. Id. If a fire was discovered that was large enough to overwhelm the fire suppression/detection systems, NFS staff would implement the Emergency Response Plan long before the public or the environment could be affected at the high (unmitigated) levels identified in the ISA

Summary. Id. The Plan includes the potential for activating local emergency response groups to respond to a fire. Id.

Finally, as with the other chemical accident sequences, the consequence assessments were based on several generally applied assumptions that yield conservative exposure estimates: (1) spills occurring when tanks are full; (2) spill amounts not limited by IROFS; (3) spill containment and Spill Response Plan not credited; (4) conservative atmospheric dispersion assumptions; and (5) exposed individual at the site boundary for one hour. Id. Use of more realistic assumptions would significantly reduce the estimated consequences for the chemical accident scenarios and hence the probability of a high consequence event. Id. In conclusion, as with the other chemical accident scenarios, the effect of all of these conservative assumptions is that the probability that significant environmental impacts would result from these accident sequences is extremely low and is significantly lower than the 10^{-5} per year suggested by the ISA Controlled Likelihood Indices alone. Id.

c) Process-Specific Scenarios

The remaining BLEU Project accident sequence evaluations identified in the Intervenor's Presentation would require specific process upsets to occur as initiating events to begin each accident scenario in motion. Id. All scenarios would occur inside facility buildings. Id. These accidents included: glovebox enclosure explosion, leak from ammonia recovery equipment, explosion of ammonium nitrate solution, NOx release due to addition of a drum of enriched scrap material to the natural dissolver, and release of calciner off-gas resulting in release of ammonia and/or hydrogen to the room. Id. at 17-18. The ISA process used to evaluate these scenarios and determine whether they exceed the regulatory risk thresholds followed the procedures outlined above. Id. at 18. Similar to the other chemical accidents already discussed, all of these scenarios require specific process upsets to occur as initiating events, have similar likelihood indices, and have multiple process-specific IROFS assigned to them. Id. As these were deemed to be high (unmitigated) consequence events, IROFS were assigned to them to render them highly unlikely. See id. Thus, the ISA Summaries alone would suggest that these accidents have probabilities of less than 10^{-5} per year.

In addition to the IROFS assigned to each scenario in these evaluations, there are many other pieces of equipment (e.g., Central Control System) that would indicate changes in process parameters, thus alerting the operator if the process was not operating properly. Id. These indicators would allow the operator(s) to intervene and prevent or mitigate the consequences of the accident scenarios. Id.

Several other conservative assumptions also rendered the estimated probabilities of these accidents higher than they actually are:

- Building process ventilation assumed not to be available
- For the glovebox enclosure or calciner off-gas hydrogen explosions, any excess hydrogen is conservatively assumed to result in a maximum explosion
- For the ammonium nitrate explosion, a concentration greater than 1% was assumed to result in explosion when, in fact, only concentrations greater than 92% have been shown to result in detonations
- IROFS such as hydrogen gas analyzers and purge valves, high temperature and pressure indicators and interlocks, and enrichment monitors, which also serve to limit accumulations of potentially explosive materials to less than explosive levels, assumed not to be available.

Id. at 18-19. Thus, even more conservatism is provided in the analyses of these accidents than for the more general scenarios discussed above. Id. at 19.

Finally, as with the other chemical accident sequences, the consequence assessments were based on several assumptions that yield conservative exposure estimates: (1) spills occurring when tanks are full; (2) spill amounts not limited by IROFS; (3) spill containment and Spill Response Plan not credited; (4) conservative atmospheric dispersion assumptions; and (5) exposed individual at the site boundary for one hour. Use of more realistic assumptions would significantly reduce the estimated consequences for these scenarios and hence the probability that a high consequence event would result. Id. at 18-19.

In conclusion, as with the other chemical accident scenarios, the effect of all of these conservative assumptions is that the probability of significant environmental impacts would result from these accident sequences is extremely low and is significantly lower than the 10^{-5} per year suggested by the ISA Controlled Likelihood Indices alone. Id. at 19.

3. Conclusions Regarding Accident Probability

It bears restating that Intervenors have overestimated the probabilities of accidents associated with the BLEU Project because they have misapplied the ISA process and have thus misinterpreted the information presented in NFS's ISA Summaries. While the Controlled Likelihood Indices assigned to certain accident scenarios may be -4 or -5, that does not mean that the probabilities of those scenarios are 10^{-4} or 10^{-5} per year. First, the likelihood indices are conservatively estimated, such that, for example, a -4 represents an order of magnitude of probability less than 10^{-4} per year. See Wheeler/Mason Decl. at 7. Second, the likelihood indices only reflect the analysis performed in the ISA in order to meet the 10 C.F.R. § 70.61 accident likelihood requirements. Frost Decl. at 2. Thus, they do not take into account (or attempt to quantify) the many unlikely events that would have to occur or the non-credited safety systems or controls that would have to fail before the accident sequences would actually occur. Id. If the conservatism of the likelihood index assessments and the non-credited unlikely events and safety systems were taken into account, the probability of a high consequence accident of any type associated with the BLEU Project would be shown to be significantly less than 10^{-5} per year.

C. Intervenors Greatly Overstate the Consequences of Potential Accidents

In addition to overstating the probabilities of occurrence of potential accidents associated with the BLEU Project, Intervenors also overstate the potential consequences of such accidents. Intervenors note that the BLEU Project ISAs assess accident sequences whose unmitigated consequences are considered "high" under 10 C.F.R. § 70.61. Intervenors' Presentation at 25. They then quote the definitions of high and intermediate consequence events from the Part 70 Statement of Considerations and imply that all such events have the potential to cause permanent injury or death to on-site workers and off-site members of the public and hence they are environmentally significant. Id. at 25-26 (quoting Proposed Rule, Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material, 64 Fed. Reg. 41,338, 41,342-43 (July 30, 1999)). They assert further that the consequences of a criticality accident associated with the BLEU Project would be as severe as or worse than the consequences of the 1999 Tokai-Mura accident in Japan. Id. at 26. Intervenors' claims are unsupported and are erroneous for several reasons.

1. Intervenors Misunderstand the Definitions of Accident Consequences

Intervenors repeatedly allege that the accident frequencies of concern for the BLEU Project include several which are "high consequence" accidents, i.e., accidents that can cause death or life-threatening injury. Intervenors' Presentation at 29. Nowhere, however, do Intervenors attempt to analyze what the consequences of those accidents would be, or discuss the various types of consequences that the regulations envision – for example, potential accident effects on plant workers onsite as opposed to potential effects on the population and environment outside the facility.

Thus, Intervenors overlook that accident consequences as defined by regulation—10 C.F.R. § 70.61—are assessed using different scales for on-site workers and off-site members of the public. A "high" consequence event to an on-site worker, such as a criticality accident, could well be a low consequence event for an off-site member of the public. Under that regulation, a "high" consequence event to an off-site member of the public is one that results in a radiation dose of greater than 25 rem or chemical exposure that could lead to serious, long-lasting health effects. 10 C.F.R. § 70.61(b). An "intermediate" consequence event is one that results in a radiation dose between 5 and 25 rem or a chemical exposure that could cause mild transient health effects. 10 C.F.R. § 70.61 (c). Thus, a high-consequence event would not necessarily pose a risk of death to an off-site member of the public and an intermediate consequence event might not pose a risk of permanent injury. By failing to even address the potential consequences of the accident, Intervenors erroneously attempt to leave the impression that the anticipated consequences of these accidents are indeed severe which, as will be shown below, is not the case.

2. Intervenors Ignore Consequence Mitigating Factors Not Credited in ISAs

Intervenors also neglect the fact, noted in the discussion of the ISA assessments and accident sequences above, that the ISA assessments of potential accident consequences were extremely conservative, particularly for chemical accidents, because in many cases the ISAs did not account for IROFS or other safety systems or controls that would mitigate the consequences of accidents assessed to be high or intermediate. For example, as discussed above, the following consequence-mitigating effects or measures were not accounted for in the ISA assessments:

- accidents occurring when tanks and process equipment were filled to less than maximum capacities;
- IROFS limiting leak and spill amounts to less than maximum process equipment capacities, e.g.:
    - operator response to a high tank level alarm that would allow intervention before large spill occurred;
    - maintenance programs ensuring that small leaks are identified and repaired before catastrophic tank or pipe failures can occur;
    - pump recirculation lines preventing pump damage and reducing the potential for catastrophic pump failure and spills of process solutions;
- containment areas for spills;
- staff responses to spills under the NFS Spill Response Plan;
- fire detection/fire suppression/automatic sprinkler systems;
- fire response under the NFS Emergency Plan;
- other realistic assumptions applicable to individual accident sequences;
- realistic atmospheric dispersion assumptions.[43]

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[43] See also Declaration of John R. Frazier Regarding the Dispersion of Airborne Effluents (Dec. 15, 2004) ("Frazier Decl."), enclosed as Attachment 3 hereto, at 5-7.

Because these uncredited mitigating factors would act to reduce the consequences of a chemical accident, the likelihood of an accident sequence defined to have "high" unmitigated consequences actually resulting in high consequences is extremely remote.

A significant further point, already mentioned above, which must be understood when evaluating the potential consequences to members of the public from a chemical accident associated with the BLEU Project, is that all of the ISA analyses assumed that the exposed individual was at the NFS property line and that the individual was continuously exposed to the chemical for one hour. Wheeler/Mason Decl. at 4. Assessments made assuming that people were exposed at realistic locations and for realistic times would have reduced exposures by more than a factor of 10 and would have reduced the potential consequences to members of the public for many if not all possible accidents below a level where serious health effects would occur. Id.

Finally, NFS performed a worst-case bounding assessment of the consequences of a chemical release at facilities associated with the BLEU Project (and elsewhere at the NFS site). Id. The analysis assumed that an accidental release would involve the entire inventory of all chemicals present at BLEU Project facilities (in any significant quantity) at any time. Id. The release was non-mechanistically assumed to be immediate and entirely unmitigated. Id. The concentration of chemicals to which an off-site member of the public could be exposed was calculated using the same method as used in the ISA process (i.e., it used conservative atmospheric dispersion assumptions and assumed that the individual would be located at the site boundary for one hour). Id. In no case did a chemical release, even under these most extreme conditions, result in an off-site fatality. Id. Therefore, while the ISA assessed the unmitigated consequences of several chemical release accident sequences to be "high," no such accident, even unmitigated, would result in a public fatality. Id.

3. Intervenors Grossly Overstate the Potential Off-site Consequences of Criticality Accidents

Intervenors also grossly overstate the potential off-site consequences of a criticality accident at NFS. They assert that those consequences are well known as a result of the accident at the Tokai-Mura facility in Japan. Intervenors' Presentation at 26. They claim regarding that accident that (1) over 400 people off-site received radiation doses "in excess of NRC standards for public exposures" (citing 10 C.F.R. §§ 20.1301, 20.1302); (2) "exposures would have been greater if the accident had not been brought under control;" (3) "the consequences would have been greater if the accident had involved HEU;" and (4) "[e]conomic damages were estimated at over $93 million." Id. at 27. However, as discussed in detail in the attached Declaration of Robert L. Frost and John R. Frazier Regarding Intervenor's Claims of Consequences From the Tokai-Mura, Japan Criticality Accident (Dec. 15, 2004) ("Frost/Frazier Decl.") (enclosed as Attachment 4 hereto), the Intervenors' claims are wrong or irrelevant and overstate the potential consequences of a criticality accident at NFS.

First, 400 off-site people did not receive radiation doses in excess of NRC limits at Tokai-Mura.[44] The NRC assessment of the consequences of the Tokai-Mura accident[45]

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[44] The Tokai-Mura accident occurred when workers inadvertently poured intermediate enriched uranium solution into a non-favorable geometry vessel. Two workers involved were killed in the accident. Frost/Frazier Decl. at 2.

[45] Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, *NRC Review of the Tokai-Mura Criticality Accident* (April 2000) ("NRC Report"), appended as Attachment 1 to SECY-00-0085, Memorandum to the Commissioners from William D. Travers, Executive Director for Operations (April 12, 2000).

reports the radiation doses received, separating recipients into six different groups of people. Four of these groups (approximately 227 people) consisted of plant workers located onsite when the accident occurred; only one group (approximately 207 people) consisted of off-site members of the public. Frost/Frazier Decl. at 3 (citing NRC Report, Fig. 7). The great majority of the off-site members of the public (approximately 180 out of 207 people) received doses not exceeding 5 mSv (0.5 rem). The remaining (approximately 27) members of the public received doses greater than 0.5 rem but not exceeding 2.5 rem. Id.

The NRC accident dose limit for off-site members of the public for fuel cycle facilities is established by 10 C.F.R. § 70.61. Accident sequences that result in doses to members of the public greater than 25 rem are deemed to have "high" consequences. 10 C.F.R. § 70.61(b)(2). Accident sequences that result in doses to members of the public greater than 5 rem but less than 25 rem are deemed to have "intermediate" consequences. 10 C.F.R. § 70.61(c)(2). Accident sequences that result in doses to offsite members of the public that do not exceed 5 rem have low consequences.[46] Therefore, because the Tokai-Mura accident did not expose any off-site members of the public to radiation doses in excess of 5 rem, it was a "low" off-site consequence event.

Second, whether exposures at Tokai-Mura would have been greater had the accident not been brought under control is irrelevant to potential consequences at NFS. The Tokai-Mura accident was not brought under control promptly, partly because emergency response at Tokai-Mura was plagued by a complete lack of planning.

[46] The regulations cited by Intervenors (10 C.F.R. §§ 20.1301, 20.1302) set dose limits for individual members of the public resulting from licensed operations—i.e., normal conditions, rather than accidents—with a maximum of 0.1 rem per year. NFS operations meet that limit.

Frost/Frazier Decl. at 3-4. The plant did not have a Criticality Accident Alarm System ("CAAS"), and there was no formal emergency plan to deal with a criticality accident. This led to "...a significant delay in development and communication of emergency protection measures for the public." Id. at 4 (quoting NRC Report at 3).

By contrast, NFS has both a CAAS and an emergency plan that comply fully with 10 C.F.R. §§ 70.22 and 70.24. If a criticality accident were to occur at NFS, the Emergency Response Organization would be aware of all pertinent site conditions and would act rapidly to bring any accident situation under control. Id. This would provide a significant measure of protection for workers and the off-site public that was not present at Tokai-Mura. Id. In addition, the NFS Emergency Response Director would notify appropriate local agencies and provide emergency response recommendations (e.g., evacuation of nearby residents, instructions to stay indoors, etc). Id. While off-site evacuation and other potential measures (e.g., traffic control) would be decided upon by local authorities and might not be necessary if the accident were quickly brought under control, such potential measures would further help to minimize accident consequences. Id. In sum, NFS has a comprehensive and effective emergency management plan to minimize consequences, in the highly unlikely event that a criticality accident were to occur. Such a plan was not in place at Tokai-Mura. Id.

Third, contrary to Intervenors' claim, the consequences at Tokai-Mura would not necessarily have been greater if the accident had involved HEU. Id. An empirical model of the effects of criticality accidents, based on data from historical accidents, relates the yield (number of fission events—which is directly related to off-site radiation dose) to the volume of fissioning material. Id. at 5. As volume increases, yield (dose) increases;

conversely, smaller volumes lead to smaller yields (doses). Id. The volume of fissile material required to achieve criticality decreases with increasing enrichment. Id. Thus, a critical volume of HEU is smaller than a critical volume of intermediate enriched uranium. Id. Therefore, in contrast to Intervenors' claim, it is most probable that a criticality accident similar to the one at Tokai-Mura, but involving HEU, would have led to lower off-site radiation doses than the Tokai-Mura event because the HEU accident would have involved a smaller volume of fissioning material. Id.

Fourth, Intervenors' statement that economic damages associated with Tokai-Mura exceeded $93 million is irrelevant to accident consequences at NFS. Id. That sum was an estimate of what plant owner JCO expected to pay in compensation to nearby residents and businesses. Id. (citing NRC Report at 2). There were no injuries, no physical damage to off-site structures, and no significant offsite contamination. Id. Thus, the $93 million does not reflect any further effects of the accident and it is not relevant to consideration of the potential consequences of a criticality accident at NFS. Id.

In summary, while a criticality accident at NFS would be highly undesirable, the off-site consequences to members of the public and the environment would be low: they would be even less than the low consequences from the Tokai-Mura accident. Id. Off-site consequences at NFS would also likely be lower than at Tokai-Mura because of the larger NFS site and the lower population density around the NFS site. Id. at 2. Thus, the Intervenors' discussion of the Tokai-Mura accident provides no basis for believing that a criticality accident at NFS would have significant consequences for either the off-site public or the environment. Id. at 5.

D. Intervenors' Characterization of BLEU Project Accident Risk as a Significant Impact Under NEPA is Wrong

As discussed above, Intervenors have misinterpreted the BLEU Project ISAs and have mischaracterized the accident probabilities associated with the BLEU Project. Intervenors have also made two further errors. They have failed to properly assess the potential consequences of potential accidents associated with the BLEU Project. Also, in the course of committing these errors, they have erroneously equated accidents that could potentially occur involving nuclear reactors or nuclear power plant spent fuel pools with the <u>much</u> less severe accidents that could potentially occur involving the BLEU Project. Therefore, Intervenors' conclusion that the NRC must prepare an EIS for the BLEU Project is also flawed from the standpoint of overall risk.

1. Intervenors Have Failed to Consider Potential Accident Consequences Along With Accident Probability in Assessing Accident Risk

Intervenors correctly state that the environmental impacts that must be considered in an EIS include "reasonably foreseeable" impacts but not impacts that are "remote and speculative." Intervenors' Presentation at 6 (citing <u>Limerick Ecology Action, Inc. v. NRC</u>, 869 F.2d 719, 745 (3d Cir. 1989)). Intervenors then argue that low probability is "key" in determining whether a particular accident scenario is remote and speculative. <u>Id.</u> (citing <u>Vermont Yankee Nuclear Power Corp.</u> (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129, 131 (1990)). They conclude their discussion by asserting that "serious accidents with a potential of one in a hundred thousand [i.e., 10^{-5}] or greater have not been ruled out by the Commission as 'remote and speculative.'" <u>Id.</u> at 8. Intervenors are correct that remote and speculative accidents need not be considered. "There is a point at which the probability of an occurrence may be so low as to render it almost totally unworthy of consideration." <u>Carolina Envtl. Study Group v. United States</u>, 510 F.2d 796, 799 (D.C. Cir. 1975). Intervenors, however, omit from their discussion the fact that potential accident consequences must also be considered in determining whether the overall risk from accidents is great enough to represent a significant (potential) environmental impact such that the NRC must prepare an EIS for the BLEU Project.

"It is undisputed that NEPA does not require consideration of remote and speculative risks." Limerick Ecology Action, 869 F.2d at 739, 745; San Luis Obispo, 751 F. 2d at 1300-01. "[R]isk . . . is generally thought of as 'the product of the probability of occurrence [and] the consequences.'" Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation), CLI-02-25, 56 NRC 340, 350 (2002) (footnote omitted). Thus, the probability of event occurrence is not the only element to the consideration of risk for NEPA purposes. Where "[i]t is only the risk of accident that might render the proposed action environmentally significant[, t]hat circumstance obliges the agency to undertake risk assessment: an estimate of both the consequences that might occur and the probability of their occurrence." City of New York, 715 F.2d at 746 (2d Cir. 1983) (emphasis added) (footnote omitted). "[I]t is entirely proper, and necessary, to consider the probabilities as well as the consequences of certain occurrences in ascertaining their environmental impact." Carolina Envtl. Study Group, 510 F.2d at 799 (emphasis added). Thus, in assessing the significance of the environmental impact potentially created by an accident, an agency is "entitled to calculate overall risk by estimating possible consequences and then discounting them by the improbability of their occurring." City of New York, 715 F.2d at 747.

The requirement to consider accident consequences as well as probability of occurrence follows from the well-established principle that "consideration of impacts must be guided by a rule of reasonableness." Limerick Ecology Action, 869 F.2d at 745 (citation omitted). In determining whether the risk of an accident is significant enough to require the preparation of an EIS, it would be entirely unreasonable to disregard the potential consequences of the accident. For example, it would make no sense to compare two accidents, both of equal probability, and say that they posed equal risk, where the first accident would cause only a small fraction of the consequences of the second. Here, it makes no sense to assess the significance of the accident risk associated with the BLEU Project without considering the fact that the consequences of potential accidents associated with the project are very small.

The Commission's most recent case law regarding the determination of the environmental significance of accident risk was established in the Vermont Yankee and Harris cases involving the amendment of reactor operating licenses to expand their spent fuel pool capacities.[47] The accident scenarios at issue in both cases were essentially the same. See Carolina Power & Light Co. (Shearon Harris Nuclear Power Plant), LBP-00-19, 52 NRC 85, 95-96 (2000) (comparing scenarios). In Harris, the Commission declined to review the licensing board's decision that the postulated scenario was remote and speculative because it had an estimated probability of 2.0×10^{-7} per year and thus the risk it posed did not require the preparation of an EIS. CLI-01-11, 53 NRC at 387-88. The Commission stated that it "need not decide here whether [the Harris intervenor's] 1.6×10^{-5} probability estimate is remote and speculative so as not to require preparation of an EIS." Id. (emphasis added). The reluctance of the Commission to establish a particular probability as a bright line threshold below which accidents are not significant and above which they are makes eminent sense when one considers that not all accidents have the same potential consequences and thus not all accidents of the same probability present the same level of risk. The decision not to draw a line is also consistent with federal case

[47] Carolina Power & Light Co. (Shearon Harris Nuclear Power Plant), CLI-01-11, 53 NRC 370 (2001); Vermont Yankee, CLI-90-7, supra; Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333 (1990).

law that states that probability and consequences must be considered to determine whether risk is environmentally significant. See City of New York, 715 F.2d at 746; Carolina Environmental Study Group, 510 F.2d at 799.

The consideration of the relative consequences of possible accidents involving the BLEU Project is critical because, as discussed below, they are much less severe than the potential consequences of the reactor and spent fuel pool accidents that were at issue in Vermont Yankee and Harris. Thus, one must consider potential accident consequences in assessing risk and determining whether the NRC must prepare an EIS for the BLEU Project.

2. Intervenors Erroneously Equate the Potential BLEU Project Accident Consequences with Potential Reactor and Spent Fuel Pool Accident Consequences

Intervenors argue that the environmental significance of the risk of accidents associated with the BLEU Project should be determined by comparing their probability with the probability of the reactor and spent fuel pool accident discussed in the Harris proceeding. See Intervenors Presentation at 7-8 (asserting based on Harris that a probability of 10^{-5} per year might not be remote and speculative); id. at 28, 32 (asserting BLEU Project accident probabilities). Intervenors' argument is erroneous because it draws from the Commission's failure to rule on whether an accident probability of 10^{-5} per year might be remote and speculative the inference that such a probability might not be remote and speculative. No such inference may be drawn from the Commission's decision. In addition, Intervenors' argument implicitly equates the consequences of an accident at the BLEU Project with those of the accident scenario at issue in Harris.

As noted, the accidents at issue in both Vermont Yankee and Harris involved a postulated scenario in which a severe reactor accident would lead to a loss of spent fuel pool cooling capability, followed by a spent fuel pool fuel cladding fire, and ultimately the release into the environment of a large quantity of radioactive material, with attendant public health and safety consequences. See Harris, LBP-00-19, 52 NRC at 95-96. The Harris licensing board described the potential radiological consequences of the accident as "serious." Harris, LBP-01-9, 53 NRC at 270 n.12 (2001) (citing Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, NUREG-1738 (Oct. 2000)).[48] In the NUREG, the NRC Staff assessed the upper bound off-site consequences to be 192 early fatalities and a total societal radiation dose of 2.37×10^7 person-rem.[49] NUREG-1738 at 3-29.[50]

By contrast, the potential consequences of a criticality accident involving the BLEU Project are much less severe than those of the reactor and spent fuel pool accidents at issue in Harris and Vermont Yankee. The consequences of the Tokai-Mura criticality accident discussed above may be used as a conservative upper bound of the consequences of a potential criticality accident at NFS. The Tokai-Mura accident resulted in no off-site

[48] The Vermont Yankee and Harris cases concerned operating reactors and spent fuel pools and NUREG-1738 concerns shutdown reactors. However, NUREG-1738 is relevant to operating reactors if one considers the scenarios in which the reactor has been shutdown for the shortest period of time (i.e., 30 days).

[49] Total societal radiation dose governs the potential number of latent cancer fatalities. See NUREG-1738 at 3-31. No conversion factor was given, but based on the societal dose and latent fatality figures reported in the appendices, the relationship is one latent fatality for every 1,140 to 2,000 person-rem. See id., Appendix 4A at 2. Thus, a total dose of 2.37×10^7 person-rem could be expected to result in 11,850 to 20,789 latent fatalities.

[50] Upper bound consequences assumed a high ruthenium source term, late evacuation, and that the reactor had been shut down for only 30 days (i.e., the newest spent fuel had only aged for 30 days). Id. Less conservative assumptions produced significantly lower consequence estimates. See id. at 3-30 (societal dose for reactor shut down for 30 days equal to 4.12×10^6 person-rem for early evacuation, which would correspond to roughly 2,060 to 3,614 latent fatalities).

fatalities. According to the NRC's assessment, approximately 180 members of the public received doses of less than 0.5 rem and approximately 27 received doses between 0.5 and 2.5 rem in the accident. Frost/Frazier Decl. at 3 (citing NRC Report Fig. 7). If the 180 people are all assumed to have received 0.5 rem and the 27 are all assumed to have received 2.5 rem, a conservative estimate of the total radiation dose to off-site members of the public would be 157.5 person-rem. That is five orders of magnitude lower than the public dose estimate for the potential accident considered in Harris and Vermont Yankee.

The potential public health consequences of a chemical accident associated with the BLEU Project are also much less than the public health consequences associated with the postulated reactor and spent fuel pool accident at issue in Vermont Yankee and Harris. As discussed above, an non-mechanistic, worst-case chemical release associated with BLEU Project facilities that assumes no mitigation by any safety systems, would result in no off-site fatalities. Further, even accident scenarios whose consequences the ISA assessed to be high would be mitigated by many factors (not credited in the ISA) so as to likely reduce their consequences to a level at which no serious health effects among members of the public would occur. Therefore, the results of a chemical accident associated with the BLEU Project would have much less effect on the off-site public than would the postulated reactor and spent fuel pool accident scenario at issue in Vermont Yankee and Harris.

The conclusion that potential accident consequences associated with the BLEU Project are much less severe than the potential accident consequences associated with reactors and spent fuel pools is entirely consistent with the Commission's understanding upon which it bases its emergency planning requirements. The Commission recognized that emergencies at fuel cycle facilities "would involve small (not life-threatening) doses, small areas, and small numbers of people. The potential risks are much lower than the risks from accidents involving chemical plants or the shipping of hazardous chemicals...."[51] Similarly, nuclear power plant emergency plans must provide for an accident classification of "general emergency" in which "there is a possibility of very large releases that could cause acute radiation effects miles from the plant. [But] [n]either releases nor doses of those magnitudes could result from accidents at fuel cycle or other radioactive material licensees." Id. at 14,054. Thus, fuel cycle facilities do not have emergency planning zones. Id. at 14,057. Evacuation planning is not required for them. Id. at 14,052. Nor are States expected to have specific emergency plans for specific fuel cycle facilities.[52]

Therefore, it would be gross error to assess the environmental significance of the accident risk associated with the BLEU Project simply by comparing its accident likelihood with the accident probabilities discussed in Harris and Vermont Yankee. Rather, one must also account for the fact that the potential consequences associated with the BLEU Project are much lower than the potential consequences of the accident at issue in those cases. Intervenors failed to do so.

3. The Risk From Potential BLEU Project Accidents Is Not Environmentally Significant

The foregoing discussion of the probability and potential consequences of accidents associated with the BLEU Project can be taken one final step forward by

[51] Emergency Preparedness for Fuel Cycle & Other Radioactive Material Licensees, Final Rule, 54 Fed. Reg. 14,051, 14,057 (Apr. 7, 1989).
[52] Emergency Preparedness for Fuel Cycle & Other Radioactive Material Licensees, Proposed Rule, 52 Fed. Reg. 12,921, 12,923 (Apr. 20, 1989).

comparing the risk from potential BLEU Project accidents with the risk found in Harris not to be environmentally significant.[53] As will be seen next, such a comparison demonstrates that the risk from potential BLEU Project accidents is also not environmentally significant. In Harris, the licensing board found (and the Commission declined to review) that the risk associated with the postulated spent fuel pool fire accident did not require the NRC to prepare an EIS. Harris, CLI-01-11, 53 NRC at 387-88. There, the accident probability was found to be 2.0 x 10^{-7} per year. Id. The upper bound off-site consequences, as assessed by the NRC Staff, were found to be 192 prompt fatalities, plus a societal radiation dose of 2.37 x 10^{7} person-rem, which could potentially give rise to 10,000 to 20,000 latent cancer fatalities. NUREG-1738 at 3-29.[54] Thus, the off-site risk from the postulated Harris accident can be expressed as the probability (2.0 x 10^{-7} per year), times the consequences (10,000 to 20,000 fatalities), or 0.002 to 0.004 fatalities per year.[55] Therefore, if the off-site risk associated with the BLEU Project is of that magnitude or less, it should similarly be found to be environmentally insignificant.

Based on the discussion of potential accidents associated with the BLEU Project above, the probability of a "high" consequence accident (as defined in 10 C.F.R. § 70.61) is well below 10^{-5} per year. As also discussed above, no potential accidents of any kind associated with the BLEU Project would result in off-site fatalities. Thus, the estimated risk of off-site fatalities is zero. The risk from the BLEU Project is thus much lower than

[53] As noted earlier, Intervenors' sole discussion of the environmental consequences of BLEU Project accidents consists of broadly labeling them as "high consequence" events.

[54] Lower bound off-site consequences would give rise to an estimated 2,000 to 3,600 latent fatalities. Note 51, supra.

[55] The measure of risk is not an estimated annual rate at which fatalities would occur. As is the case with all very low probability events, the events are expected never to occur. The expression of risk in terms of fatalities per year is simply a means of comparing the risks posed by different accident scenarios.

the risk associated with the accident at issue in Harris, and is thus insufficient to require the NRC to prepare an EIS.[56]

Moreover, if societal radiation doses rather than off-site fatalities are used as the metric of accident consequences, the risk from the BLEU Project still remains insignificant. (For purposes of making this comparison, one must focus on a criticality accident at the BLEU Project, for that type of accident is the only one that will result in societal radiation doses.) As noted above, the off-site dose estimated for the postulated Harris accident was 2.37×10^{7} person-rem. With an accident probability of 2.0×10^{-7} per year, that gives rise to an off-site societal radiation dose risk of approximately 4.7 person-rem per year. By contrast, the off-site dose from the Tokai-Mura accident, which can be taken as a conservative upper bound for a criticality accident at the BLEU Project, was approximately 157.5 person-rem. With a BLEU Project accident probability of less than 10^{-5} per year, that gives rise to a societal radiation dose risk of less than 1.6×10^{-3} person-rem per year. Again, this risk is three orders of magnitude lower than the corresponding risk posed by the postulated accident in Harris. Thus, the risk posed by a criticality accident at the BLEU Project does not require the NRC to prepare an EIS.

Therefore, based on the metrics of potential off-site fatalities, potential total off-site societal radiation doses, and potential chemical injury, the accident risk from the BLEU Project is orders of magnitude less than the risk from the postulated accident in Harris. Thus, because the risk in Harris was found not to require the preparation of an

[56] Even if, as an extra measure of conservatism, the on-site fatalities at Tokai-Mura (two), were added to the zero off-site fatalities, the total risk of the BLEU Project would be less than 2.0×10^{-5} fatalities per year, which would still be at least two orders of magnitude less than the off-site risk found insufficient to require an EIS in Harris.

EIS, the accident risk from the BLEU Project is far below the level of significance that would require the preparation of an EIS.

E. The BLEU Project Meets None of the NRC Staff's Considerations for Finding Significant Environmental Impacts

NUREG-1748 provides guidance for the NRC Staff to meet its statutory and regulatory requirements in conducting NEPA evaluations for Nuclear Material Safety and Safeguards ("NMSS") actions. NUREG-1748 at 1-1. The guidance offers provides ten considerations for the NRC Staff to use in evaluating whether the proposed action's impacts are significant. NUREG-1748 at 3-12–13. The guidance provides that if the project fits any of the considerations, "then an EIS is normally required." Id. at 3-13.

Intervenors argue that the environmental impacts of the proposed BLEU Project meet three of the ten significance considerations discussed in NUREG-1748 and, therefore, claim that an EIS is warranted. Intervenors' Presentation at 32-33. Intervenors thus admit the BLEU Project does not meet the other seven considerations, which need not be addressed here.

Regarding the three that they raise – that the BLEU Project (1) "would inflict 'undesirable public health or safety effects'" by posing a high public health risk to the nearby community; (2) possesses several "'unique geographical characteristics'" by being situated in a long narrow valley and next to the Nolichucky River; and (3) has environmental impacts that are "highly uncertain' and involve 'unknown risks,'" – Intervenors' arguments are unsound as well as unsupported. Intervenors fail to point to any inconsistency between the NRC Staff's NEPA analysis and the guidance in NUREG-1748, but only assert disagreement with the NRC Staff's findings of no significant impact. Such a disagreement, without more, is no grounds for overturning the NRC Staff's determination that an EIS need not be prepared. 10 C.F.R. § 51.25; Pacific Gas & Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-86-12, 24 NRC 1, 8 (1986); Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation), LBP-03-30, 58 NRC 454, 470 (2003). The following examination of the three NUREG-1748 considerations asserted by the Intervenors demonstrates that the BLEU Project does not meet them.

Consideration 1: Are there undesirable health or safety effects?

The NRC Staff's NEPA review for the BLEU Project identified no undesirable health or safety effects resulting from the proposed BLEU Project activities. Intervenors argue the BLEU Project "poses a relatively high public health risk to a large community of people," Intervenors' Presentation at 33-34.[57] However, the NRC Staff's NEPA analysis discussed in detail potential impacts to the site's surrounding community, and concluded otherwise.

The NRC Staff recognized that BLEU Project operations might require modification to NFS's existing air effluent permit because of an increase in nitrogen oxide emissions. 1st EA at 2-10, 5-11. In addition, the NRC Staff also expected a negligible increase in radiological effluents. Id. at 5-10. However, NFS would be required to comply with existing and new environmental permits set by state authorities to ensure any increase is within acceptable limits, along with NRC's radiological release

[57] While Intervenors do not clearly explain what "high public risk" they are referring to, it is clear from the rest of their presentation that they are citing the alleged public health risks posed by accidents. As noted earlier, Intervenors raise no issue withy respect to normal BLEU Project operations.

limits. Id. at 5-11. Thus, the NRC Staff concluded that "in relation to existing plant impacts, and in light of existing regulatory controls, these impacts represent a small change to existing conditions in the area surrounding the plant." Id. Therefore, no undesirable health or safety effects are expected.

Moreover, each of the EAs prepared and the three FONSIs issued for the BLEU project discussed potential impacts to the environment, workers, and public health and safety under accident conditions; each FONSI concluded that sufficient safety controls were in place to ensure that operations would be conducted safely. 67 Fed. Reg. at 45,557 (July 9, 2002); 68 Fed. Reg. at 61,237 (Oct. 27, 2003); 69 Fed. Reg. 34,200-01 (June 18, 2004). Thus, the NRC Staff determined that the proposed BLEU Project did not pose a risk of "significant impact" such that an EIS was required.

All that Intervenors offer in response to the NRC Staff's analyses is the bald assertion that, because "an offsite radiological and/or chemical release could affect the health and safety of many people...[a]n EIS should be prepared to address these impacts." Intervenors' Presentation at 34 (emphasis added). Intervenors have provided no link between the theoretical possibility that an accidental release could affect public health and the actual likelihood that such an effect would take place – a likelihood that, as discussed in the preceding sections, is remote. Such unsupported opinion provides no basis for overturning the NRC Staff's determinations.[58]

[58] If taken literally, Intervenors' argument would lead to the absurd result that any federal action that could theoretically result in adverse public health consequences would require the preparation of an EIS.

Consideration 2: Are there unique characteristics of the geographic area, such as proximity to historic or cultural resources, park lands, prime farmlands, wetlands, wild/scenic rivers, or ecologically critical areas?

The NFS Erwin, TN location and its surrounding environment have been thoroughly examined several times and found not to meet the significance standard set forth by this consideration. Intervenors claim that it meets the standard because the site is located in a long, narrow valley which allegedly could trap airborne radiological and chemical releases, and that the site is situated next to the Nolichucky River, which is a source of recreation and drinking water for the nearby population. Intervenors' Presentation at 35. Their unsupported and insufficient arguments must be rejected.

The NFS site was identified and analyzed, including an evaluation of all potentially affected environments, in the DOE EIS. See DOE EIS at 2-12. This review concluded that no significant impact to the environment would result. Id. at 2-36. The License Renewal EA evaluated environmental impacts to the surrounding location from continued operations. 64 Fed. Reg. 5,681. The NRC Staff then conducted a comprehensive analysis of the location for the BLEU Project license amendments, 1st EA at 3-1–3-19, and reaffirmed that analysis in the 2nd and 3rd EAs. 68 Fed. Reg. 61,236-37, 69 Fed. Reg. at 34,200.

The 1st EA determined that impacts from the BLEU Project would be confined to modification or construction of new facilities on the NFS site. 1st EA at 5-3. Thus, impacts to cultural resources would be no different to those evaluated under the License Renewal EA. Id. No historical structures or Native American sacred sites were known to exist on the NFS site, and regional historical properties will not be disturbed. Id. at 3-6, 5-3. No prime farmland was identified near the site. Id. at 3-5. No park land was identified that had previously not been disturbed. No wetlands would be destroyed because of the BLEU Project. Wetlands that were being eliminated from ongoing remediation activities unrelated to the BLEU Project were being replaced by increasing another wetland on site. Id. at 3-7. No scenic or wild rivers were identified. In this regard, the 1st EA described the Nolichucky River as a "typical river of eastern Tennessee." Id. at 3-11. No critical ecological areas were identified. Therefore, the NFS site does not meet the standard set forth by Consideration 2.

Intervenors have provided no factual support for their assertion that there is the potential for chemical and radiological airborne releases to be trapped in the valley where the BLEU Project is to be located. Intervenors' bare assertion falls far short of meeting the Intervenors' burden to put forth credible evidence contradicting the NRC Staff's findings and demonstrating that operation of the facility might pose a significant environmental impact. White Mesa, LBP-02-19, 56 NRC at 123.

Moreover, it is highly unlikely that consequences from chemical or nuclear criticality accidents would have any serious offsite impact, thus Intervenors' concern that chemical and radiological releases might somehow become trapped in the valley is unwarranted. No "trapping" of airborne releases in or near the NFS site would occur, because representative, site-specific meteorological data confirm that adequate and sustained atmospheric dispersion conditions are present at the NFS site throughout the year. See Frazier Decl. at 3-5. In addition, the atmospheric modeling performed in the NFS ISA process employed conservative assumptions for any atmospheric dispersions.

Id. at 5. Thus, the potential offsite exposure from any accident has been overestimated rather than overlooked.

Intervenors also assert that the EA does not address the environmental impacts of accidental releases on the Nolichucky River. Intervenors' Presentation at 35. This argument must also be rejected. First, with regard to the NUREG-1748 consideration's standard, Intervenors have failed to provide any basis to conclude that the Nolichucky River fits the definition of a "unique" geographical characteristic surrounding the site. Intervenors do not assert that the Nolichucky is wild or scenic as the NUREG-1748 consideration requires. NUREG-1748 at 3-12. Indeed, the NRC Staff described the Nolichucky is a "typical river of eastern Tennessee." 1st EA at 3-11. Yet, Intervenors assert only that because the river is a source of drinking water and recreation, and EIS needs to be prepared to discuss the environmental impacts of potential accidents on the "people and wildlife that depend on" the river. Intervenors' Presentation at 35.

However, the NRC Staff's NEPA analysis discussed non-radiological and radiological impacts from normal operations, including those to the Nolichucky River, and concluded that discharges would not significantly impact the river because they would be diluted in the water. 1st EA at 5-2. Further, the NRC Staff's NEPA analysis thoroughly discussed the potential for accidents at the BLEU facility. Id. at 5-7–5-10. Each of the NRC Staff's FONSIs recognized the potential for accidents to occur that could result in chemical or radiological releases to the environment, including the river. See 67 Fed. Reg. at 45,557 (July 9, 2002); 68 Fed. Reg. at 61,237 (Oct. 27, 2003); 69 Fed. Reg. 34,200-01 (June 18, 2004). The NRC Staff concluded that sufficient controls were in place to ensure safe facility operations. Id. Therefore, the risk presented by

BLEU Project Operations did not amount to a significant impact on the surrounding environment. Id. Intervenors have failed to provide any basis to conclude otherwise.

Consideration 3: Are the impacts on the human environment highly uncertain, or do they involve unique or unknown risks?

The 1st EA sets forth expected environmental impacts from normal operations for the BLEU Project. 1st EA at 5-1–5-7. These impacts are not uncertain and are not contested by Intervenors. Any uncertainty would lay in the potential for accidents and the resultant consequences. The NRC Staff was well aware of a potential result from an accident – the release of larger concentrations (compared to normal operations) of chemical and/or radiological constituents over a shorter period of time. 1st EA at 5-1. The NRC Staff concluded in its accident analyses that the processes to be carried out in the BLEU project "will function safely with no significant adverse impacts to safety or the environment." Id. at 5-7–10; see also 67 Fed. Reg. at 45,557; 68 Fed. Reg. at 61,237; 69 Fed. Reg. at 34,200-01. Thus, the NRC Staff concluded there was little if any uncertainty with regard to potential impacts from potential accidents. See 1st EA at 5-7–10.

Intervenors claim uncertainty exists because IROFS, management measures, and programmatic commitments must be "properly implemented." Intervenors' Presentation at 36 (citing UNB Amendment SER). However, NFS has committed to establishing management measures to maintain the availability and reliability of the IROFS. See UNB Amendment SER (July 2003) at § 15; BLEU Preparation Facility SER (Jan. 2004) at § 16; OCB/EPB SER (July 2004) at § 16. For example, the OCB/EPB SER states that the commitments for IROFS management measures contained in NFS's license amendment application have been incorporated into NFS's license. OCB/EPB SER at § 16.0. Eight management measure areas exist: configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other QA elements. Id. The NRC Staff concluded that NFS had met the applicable requirements under 10 C.F.R. Part 70 for each of the eight areas. Id. at §§ 16.1.1-8. Intervenors have provided no factual or legal basis upon which to question whether these measures will be properly implemented.

In addition, Intervenors claim that uncertainty exists in the level of judgment employed in the ISAs because the Initiating Event Frequency Index and the IROFS Effectiveness of Protection Index were developed based on "past experience, engineering judgement [sic], analytical data, industry acceptable values, and/or any other applicable information." Intervenors' Presentation at 36. The claim is baseless. The information used in the preparation of the ISAs is exactly the type of information approved of by the Commission for the NRC Staff's NEPA analysis. See Harris, LBP-01-9, 53 NRC at 252. An NRC Staff determination to issue a FONSI rather than prepare an EIS can be "based on existing materials available to it, probabilistic and otherwise, supplemented by additional information it might obtain from the Applicant in an environmental report or through requests for additional information." Id. For example, with regard to the processing facilities to be used in the BLEU Project, the NRC noted that "[m]any of the proposed process operations are patterned after existing, NRC licensed processes, so operational experience and history build confidence that operations can be executed safely." 1st EA at 5-7. Therefore, any NRC Staff finding of no significant impact based on such material is sufficiently "certain" to satisfy Commission requirement.

Finally, as explained in the above discussion, the overall risk presented by BLEU Project operations is very low. The NRC Staff concluded – and Intervenors' have failed to dispute – that NFS has met its Part 70 burden to limit the risk of potential high and intermediate consequence accidents below the regulatory thresholds. See UNB Amendment SER (July 2003) at 94; BLEU Preparation Facility SER (Jan. 2004) at 21.0-1; OCB/EPB SER (July 2004) at 68. Furthermore, NFS employed conservative assumptions in its ISAs such that potential accident probabilities and consequences are much lower than the ISA Summaries indicate. See discussion in Sections III.B. and C., supra. Therefore, no "uncertainty" exists as to the environmental impacts of the BLEU Project.

In sum, the BLEU Project meets none of the considerations for determining that a significant environmental impact potentially exists, as set forth in NUREG-1748. Intervenors' "qualitative criteria" arguments amount at most to a disagreement with the NRC Staff's ultimate conclusion that it was not necessary to prepare an EIS. Such a disagreement provides no basis to alter the NRC Staff's finding of no significant impact. Therefore, the NRC Staff justifiably concluded that the BLEU Project would have no significant impact on the human environment, and that no EIS was warranted.

F. Intervenors Suggestions for EIS Discussion Topics Fail to Provide a Substantive or Specific Basis Warranting the Preparation of an EIS

Intervenors argue that an EIS should be prepared to remedy alleged shortcomings in the NRC Staff's NEPA evaluations. Intervenors' Presentation at 37-39. Intervenors claim, without citing specifics or providing supporting analysis, that the NRC Staff's EAs provided only a "cursory" description of the BLEU Project's impacts from operations and failed to relate accidents to the surrounding environment. Intervenors' Presentation at 37. There is no merit to these arguments. To the contrary, the NRC Staff's environmental evaluations thoroughly discussed the expected impacts from normal operations to all surrounding environments. See discussion supra Section II.B. The NRC Staff concluded that expected radiological and nonradiological effluents to the air and water represented a "small change to existing conditions in the area surrounding the plant." 1st EA at 5-11. Moreover, the NRC Staff's environmental evaluations considered impacts from potential accidents and concluded that those potential impacts were not significant. Id. at 5-10–11.

Intervenors find fault in the BLEU Project EAs for allegedly failing to consider more than the no-action alternative. Intervenors' Presentation at 38. This argument is also meritless. First, Commission regulations provide that the NRC Staff may incorporate material by reference into its environmental analysis to eliminate repetitive discussion or analysis of issues. 10 C.F.R. Part 51 App. A at 1(b). The NRC Staff's environmental review stated that alternatives to the proposed action had been sufficiently considered in the DOE EIS. 1st EA at 2-1. The DOE EIS did discuss the environmental impacts for a range of alternatives, namely an HEU downblending operation at four different sites, including NFS's Erwin, TN site, and three different downblending processes, with four variations in the type of end-product. DOE EIS at 1-4, 1-6. Thus, the NRC Staff did consider more than the no-action alternative.

In any event, the implication that it is insufficient to consider only the no action alternative is erroneous. An EA need only consider the no action alternative to the licensing action. NUREG-1748 at 3-5. Indeed, "for actions having a very small impact, it is reasonable to consider a limited range of alternatives." Id.; see also 10 C.F.R. § 51.30(a)(iii) (the EA must include "the environmental impacts of the proposed action and alternatives as appropriate.") (emphasis added). Moreover, Intervenors have failed to point to any particular alternative that the NRC staff did not consider that it should have.

Intervenors' comparison to the Draft EIS the NRC prepared for DOE's planned MOX fuel facility is inapposite. Intervenors' Presentation at 38 n.20. Since the BLEU Project did not require the preparation of an EIS, the only required determination was if the environmental impacts from the facility were significant. 10 C.F.R. § 51.20(b)(14). The NRC Staff concluded they were not, and Intervenors have failed to provide any evidence otherwise.

Finally, Intervenors argue that they are "entitled to the benefit of expert review and analysis" that public circulation of a draft and final EIS would provide. Intervenors' Presentation at 39. Intervenors' have not pointed to any deficiency in the NRC Staff's NEPA review that would be remedied by further public circulation. This argument, too, must be rejected.

IV. CONCLUSION

As the above discussion amply demonstrates, the NRC Staff has assessed the expected and potential environmental impacts from the BLEU Project backwards and forwards several times. In its review of the proposed BLEU Project, the NRC Staff drew upon the environmental evaluations contained in both the DOE EIS and the License Renewal EA. The NRC Staff carefully considered what information contained in those reports was applicable to the proposed project. Specifically, the NRC Staff recognized that a range of alternative actions had been considered in the DOE EIS. In addition, the NRC Staff recognized that several of the processes to be employed at the BLEU site had either already been approved and conducted at the site or had been approved under other NRC licenses. All that remained was an evaluation of the environmental impacts resulting from new BLEU Project activities. That is precisely what the NRC Staff did.

The NRC Staff evaluated the environmental impacts for the entire project in the 1st EA. It subsequently prepared two additional EAs that were based on up-to-date information, over the course of nearly two years, provided by NFS. Ultimately, the NRC Staff concluded that no significant environmental impacts would result from the proposed action and no EIS need be prepared, and issued three FONSIs. The NRC Staff's EAs showed that all the considerations identified in the NRC Staff's own guidance for determining the significance of environmental impacts pointed to no significant impacts. The NRC Staff's conclusions are fully justified.

In sum, the BLEU Project expected and potential environmental impacts have undergone extensive environmental review. The NRC Staff has fulfilled its NEPA responsibilities by taking a hard look at those expected and potential impacts and found that none to be significant. Intervenors' arguments to the contrary have failed to identify any deficiency in the NRC Staff's NEPA review, and have fallen far short of raising issues warranting the preparation of an EIS. Intervenors' arguments should be rejected, and the NRC Staff's decision to issue a FONSI for each license amendment should be upheld.

Respectfully submitted,

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Dated: December 22, 2004

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing "Applicant's Written Presentation in Response to Intervenors' Legal and Evidentiary Presentation" and "Notice of Appearance" for Mr. Matias F. Travieso-Diaz and "Notice of Appearance" for Mr. Timothy J. V. Walsh were served on the persons listed below by electronic mail or by facsimile and deposit in the U.S. mail, first class, postage prepaid, this 22nd day of December, 2004.

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Timothy J V Walsh

Timothy J. V. Walsh

ATTACHMENT 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Presiding Officer

In the Matter of	)	
	)	Docket No. 70-143
Nuclear Fuel Services, Inc.	)	Special Nuclear Material
	)	License No. SNM-124
(Blended Low Enriched Uranium Project)	)	

Declaration of Jennifer K. Wheeler and Carol L. Mason
Regarding Chemical Accident and Risk Issues

Jennifer K. Wheeler and Carol L. Mason state as follows under penalty of perjury:

I. Discussion of ISA Process

Integrated Safety Analysis (ISA) is "*a systematic analysis used to identify facility and external hazards and their potential for initiating accident sequences, the potential accident sequences, their likelihood and consequences, and the Items Relied on For Safety (IROFS). Integrated means joint consideration of, and protection from, all relevant hazards, including radiological, nuclear criticality, fire, and chemical. However, with respect to compliance with the regulations of this part, the NRC requirement is limited to consideration of hazards directly associated with NRC licensed radioactive material*" (from 10 CFR 70.4, definition of Integrated Safety Analysis). An **ISA Summary** is a document (or documents) that provides a synopsis of the results of the ISA and contains the information specified in 10 CFR 70.65(b). Guidance for developing and documenting ISAs is provided in NUREG-1513, *Integrated Safety Analysis Guidance Document*, and NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*.

The first step in performing the ISA is to conduct a **Process Hazards Analysis** (PHA). A PHA is a method used to identify credible accident sequences/scenarios resulting from a single upset event and the controls needed to prevent or limit their occurrence or mitigate their potential consequences. Accident sequences/scenarios are defined as a series of unintended events, which if left unmitigated would have a negative safety impact. Identification of possible accident scenarios is accomplished through team analysis, whereby each process area is systematically evaluated to determine the potential impact of specific component failures. The ISA Team evaluates all types of failures for each valve, tank, pipe or control system identified on the system design drawings, and each credible failure mode for each component is identified as a specific accident sequence. The ISA Team consists of members who have a variety of expertise and experience in safety, engineering, operations, or maintenance. Process safety information used by the team consists of hazardous material properties (e.g., Material Safety Data Sheets [MSDS]), process flow diagrams, engineering design packages, and process equipment information. Hazards associated with each process are identified (i.e., radioactive and fissile materials, chemical and facility hazards, flammable and toxic materials, hazardous reactions, and use/storage locations).

After all credible accident sequences are identified by the PHA, a **Consequence Analysis** is performed for each accident sequence. Consequence analysis is a method used to determine whether accident sequences/scenarios defined in the PHA have the potential to exceed the Intermediate or High exposure levels prescribed in 10 CFR 70.61 (b) and (c). In the ISA process, consequences for each accident scenario are determined without crediting engineered or administrative controls (unmitigated consequences), thus conservatively producing an estimate of the worst-case scenario outcome. For example, specific mitigating features such as containment dikes are not credited when determining unmitigated consequences.

The sequences/scenarios are then grouped by each safety discipline (i.e., radiological, chemical, fire) into bounding accident sequences – accident scenarios of a similar type where all failure modes result in consequences within the same consequence category (all high, all intermediate, or all low). Because Nuclear Criticality accidents are considered high consequence events by definition, consequence analysis is not required.

If an accident can result in a range of consequences, all possibilities must be considered, including the maximum source term and the most adverse conditions that could occur. This effectively adds to the conservative nature of these analyses by again assuming the worst-case scenario outcome occurs. Industry guidance documents such as NUREG/CR-6410, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, are used to develop methods for estimating consequence levels. If the unmitigated consequence level estimates are determined to be high or intermediate, they are reviewed with engineering and the other safety disciplines to ensure the basic assumptions are reasonably conservative. The consequence level estimates are also reviewed to determine whether facility changes are warranted to improve the consequence category or the conditions in which the accident is credible. The Consequence Analysis documents are reviewed by an independent peer reviewer to verify that the bounding accident sequences appropriately bound the individual PHA scenarios such that the low consequence events do not have the potential to exceed intermediate levels, and intermediate consequence events do not have the potential to exceed high levels as defined in 10 CFR 70.61 (b) and (c).

After the Consequence Analysis is performed, a **Risk Assessment** is performed for each accident scenario previously identified as having high or intermediate consequences. Risk Assessment is a method for categorizing accident sequences in terms of their likelihood of occurrence and their consequences of concern. The purpose of the qualitative risk index method is to identify which accident sequences have consequences, as estimated in the Consequence Analysis or assumed for Nuclear Criticality accidents, that exceed the performance requirements of 10 CFR 70.61. Under 10 CFR 70.61, credible sequences/scenarios that result in High or Intermediate consequences must have their likelihood or consequences reduced so that High consequence scenarios are no more than highly unlikely and Intermediate consequence scenarios are no more than unlikely. Therefore, those sequences/scenarios require designation of IROFS and supporting management measures to reduce their risk. Risk Assessment methods are based on the example presented in Appendix A to Chapter 3 of NUREG-1520.

Each credible accident scenario is assigned an unmitigated, uncontrolled consequence severity category as defined in Table 1 below.

Table 1 – Radiological and Chemical Consequence Severity Categories Based on 10 CFR 70.61

	Workers	Offsite Public	Environment
Consequence Category 3 (High Consequence)	TEDE[1] ≥ 100 rem ≥ ERPG3[2]	TEDE ≥ 25 rem ≥ 30 mg soluble uranium intake ≥ ERPG2	
Consequence Category 2 (Intermediate Consequence)	25 rem ≤ TEDE < 100 rem ≥ ERPG2 but < ERPG3	5 rem ≤ TEDE < 25 rem ≥ ERPG1 but < ERPG2	Radioactive release averaged over a 24 hour period of > 5000 x Table 2, Appendix B of 10 CFR 20
Consequence Category 1 (Low Consequence)	Accidents of lesser radiological and chemical exposures than those listed above in this column	Accidents of lesser radiological and chemical exposures than those listed above in this column	Radioactive releases producing effects less than those listed above in this column

The chemical consequence severity of the postulated accident scenario is based on a maximum amount of hazardous material inventory present and a worst-case release pathway. The consequence categories are based on Emergency Response Planning Guidelines (ERPG) values developed by the American Industrial Hygiene Association for use in evaluating the effects of accidental chemical releases on the general public. ERPGs are estimates of concentration ranges for specific chemicals above which acute (<1 hr) exposure would be expected to lead to adverse health effects of increasing severity for ERPG-1, -2, and -3. ERPG values are defined as follows:

ERPG-3: maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hr without experiencing or developing life-threatening health effects.

ERPG-2: maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hr without experiencing or developing irreversible or other serious health effects or symptoms that could impair an individual's ability to take protective action.

ERPG-1: maximum airborne concentration below which it is believed nearly all individuals could be exposed for up to 1 hr without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.

[1] Total effective dose equivalent.
[2] Emergency Response Planning Guideline, Level 1, 2, or 3.

For the public, a high chemical consequence accident sequence is defined in Table 1 as one that results in an airborne concentration of >ERPG-2. The determination of consequence level is based solely on unmitigated chemical releases, i.e., no credit is taken for any action to reduce or limit consequences. However, as shown in the discussions in Section II below, all accident sequences identified as high or intermediate consequences have one or more controls or conservative assumptions associated with them such that the mitigated, or controlled, consequence level would be significantly lower. In addition, the ERPG values are based on a 1-hr exposure. Generally, a person would not be expected to remain at a point within a toxic chemical plume for any length of time. Thus, the assumed 1-hr exposure is conservative.

Another conservative assumption is the fact that the maximum consequences are estimated at the nearest site boundary even though the nearest offsite population is located some distance from that point. Dispersion parameters used to determine concentrations at various distances, in fact, decrease exponentially with distance. Thus, if the actual distance to a public receptor is increased by 100 m, the concentration at that point decreases by approximately a factor of 10. The nearest residence is approximately 150 m from the NFS site boundary, so this effect would significantly reduce the concentration to which members of the public would be exposed and could well reduce effects above ERPG-2 to below ERPG-2 and hence, with respect to the off-site public, reduce "high" consequence events to "intermediate" (and "intermediate" to "low").

Furthermore, NFS previously performed an analysis to show that off-site public fatalities would not result from a chemical release involving BLEU Project facilities.[3] The analysis assumed that an accidental release would involve the entire inventory of all chemicals present at BLEU Project facilities at any time. The release was non-mechanistically assumed to be immediate and entirely unmitigated. The concentration of chemicals to which an off-site member of the public could be exposed was calculated using the same method as used in the ISA process (i.e., it used conservative atmospheric dispersion assumptions and assumed that the individual would be located at the site boundary for one hour). Because there are no universally established chemical exposure levels at which fatalities might occur, the exposure level for immediate fatalities was taken for this specific analysis to be 5 times ERPG-3 levels. It should be noted that, even if a maximum unmitigated consequence from a single accident scenario were >ERPG-3, this exposure might endanger life but does not result in immediate fatalities.

The basis for the level of 5 times ERPG-3 was analogy to radiological exposure and 10 CFR 70.61 limits. The radiological exposure level at which 50% of the exposed population could die within 60 days without medical treatment is 5 times the 10 CFR 70.61 life endangerment level for radiological exposure (5 times 100 rem, or 500 rem). Therefore, it was analogously assumed that the chemical exposure level at which fatalities could occur is 5 times the 10 CFR 70.61 life endangerment level for chemical exposure (i.e., 5 times ERPG-3). This analysis was conducted for all chemicals that may be present in any significant quantity in any of the BLEU Project facilities at any given time. It showed that off-site exposures would be less than this critical exposure level. Therefore, fatalities among members of the public would not be expected as the result of any chemical accidents.

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[3] Blast Damage Analysis of MAA Facilities (U), Rev. 0, 10/21/03 (document marked **Confidential RD**, title of document Unclassified); Critical Target Area Analysis of OCB/EPB Facilities, BLEU Complex, Rev. 1, 7/22/04 (document marked **Official Use Only**).

For each credible accident sequence, the initiating event that leads to the accident is identified. If a single initiating event cannot be identified, the conditions that must be met to create the accident are analyzed. An Initiating Event Failure Frequency Index (see Table 2 below) is assigned to each accident scenario based on past operating experience, engineering analysis, analytical data, industry acceptable values, and/or any other applicable information. The Initiating Event Failure Frequency is defined as the estimated frequency of occurrence of the initiating event or initiating set of conditions (where a single initiating event cannot be identified).

Table 2 – Initiating Event Failure Frequency Index Numbers

Frequency Index	Description	Comments
-5	Not credible	If initiating event is not credible, no IROFS are needed
-4	Physically possible, but not expected to occur.	
-3	Not expected to occur during plant lifetime.	
-2	Not expected, but might occur during plant lifetime.	
-1	Expected to occur during plant lifetime.	
0	Expected to occur regularly during plant lifetime.	
1	A frequent event	

IROFS are then identified and assigned to all credible high or intermediate consequence accident scenarios. **IROFS** are structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents or to mitigate the potential consequences of accidents at a facility that could exceed the performance requirements of 10 CFR 70.61. An IROFS provides a safety function that serves to reduce the risk (likelihood and/or consequences) associated with a specific accident scenario. Each High consequence accident sequence/scenario should have at least two IROFS to mitigate or prevent the accident sequence to meet the performance requirements; however, a single (sole) IROFS may be used to mitigate/prevent an accident sequence/scenario based on an approved analysis. Sole IROFS may be relied on to mitigate or prevent an Intermediate accident sequence/scenario.

IROFS are categorized as Passive Engineered Controls, Active Engineered Controls, Enhanced Administrative Controls, or Administrative Controls. Passive Engineered Controls are devices that use only fixed, physical design features without any required human or mechanical action (i.e., tank overflows, containment dikes). Active Engineered Controls are pieces of equipment that may monitor a parameter, control flow or temperature, initiate or stop a flow, etc. (i.e., automatically actuated valves, temperature control systems). Enhanced Administrative Controls are Administrative Controls that are augmented by the addition or combination of a physical device that alerts the operator that an action is needed. Administrative Controls are procedural human actions that may include instructions or operating steps.

Each IROFS is assigned an IROFS Effectiveness of Protection Index as shown in Table 3 below. The Effectiveness of Protection Index is defined as the level of protection that the identified controls will prevent or mitigate the accidental consequence given the initiating event (or set of conditions) occurs. The Index is assigned to each IROFS based on industry accepted values, past operating experience, engineering analysis, analytical data, and/or any other applicable information.

Table 3 – IROFS Effectiveness of Protection Index

Effectiveness of Protection Index	Type of IROFS**
-4* ***	Protected by an exceptionally robust inspected passive engineered control (PEC). Exceptionally robust Management Measures to ensure availability.
-3*	Protected by an inspected single PEC <u>or</u> exceptionally robust functionally tested active engineered control (AEC) with a trained operator backup. Adequate Management Measures to ensure availability.
-2*	Protected by a single functionally tested AEC <u>or</u> protected by a trained operator performing a routine task with an approved procedure, an enhanced administrative control, or an administrative control with a large safety margin. Adequate Management Measures to ensure availability.
-1	Protected by a single administrative control <u>or</u> a trained operator performing a non-routine task with an approved procedure.
0	No protection

*Indices less than (more negative than) "-1" should not be assigned to IROFS unless the configuration management, auditing and other management measures are of high quality, because without these measures, the IROFS may be changed or not maintained.
**The index value assigned to an IROFS of a given type may be one value higher or lower than the value given. Criteria justifying assignment of the lower value should be given in the narrative describing ISA methods. Exceptions require individual justification.
***Rarely can be justified by evidence. Further, most types of single IROFS have been observed to fail.

During the development of NFS' overall risk assessment approach, NFS reviewed risk index methodologies for other fuel cycle facilities with similar complexities. The manner in which NFS adopted use of the terms and methodologies for qualitative risk assessment is fully consistent with the guidance provided by NUREG-1520 (see Appendix A to Chapter 3) and the regulatory approach used by other fuel cycle facilities.

NUREG-1520 was developed by NRC with significant input and comment from the fuel cycle facility industry and the Nuclear Energy Institute (NEI). NFS modeled Table 3 above after Table A-10 in Appendix A to Chapter 3. Table A-10 was developed through meetings between NRC, NEI, and industry based on documented industry reliability data. Data for passive engineered and active engineered controls were gleaned from facility operating histories and industry standards such as those produced by the Institute of Electrical and Electronics Engineers (IEEE).

The failure probability indices in Table A-10 are described as ranges, spread over two orders of magnitude. As shown in the table below, the IROFS Effectiveness of Protection indices that NFS employs are conservative in that the least negative indices of the range available are applied. As a result, the NFS definition of "Highly Unlikely" corresponds to a -4 highly unlikely index using the most conservative (least negative) failure probability index values. If the least conservative (most negative) failure probability index values were chosen along with an index of -5 to define "Highly Unlikely", the same safety envelope would be established. Therefore, it is appropriate to use failure probability indices within the defined highly unlikely range (i.e., -4 or -5). The table shown below provides a comparison between the values in Table A-10 from NUREG-1520 and the NFS IROFS Effectiveness of Protection Indices.

Table A-10 of NUREG-1520			NFS (see Table 3 above)
Probability Index No.	Probability of Failure on Demand	Based on Type of IROFS	IROFS Effectiveness of Protection Index
-6	10^{-6}		
-4 or -5	$10^{-4} - 10^{-5}$	Exceptionally robust passive engineered IROFS (PEC), or an inherently safe process, or two redundant IROFS more robust than simple admin. (AEC, PEC, or enhanced admin.)	-4
-3 or -4	$10^{-3} - 10^{-4}$	A single passive engineered IROFS (PEC) or an active engineered IROFS (AEC) with high availability	-3
-2 or -3	$10^{-2} - 10^{-3}$	A single active engineered IROFS, or an enhanced admin. IROFS, or an admin. IROFS for routine planned operations	-2
-1 or -2	$10^{-1} - 10^{-2}$	An admin. IROFS that must be performed in response to a rare unplanned demand	-1
			0

The process used to develop Table A-10 can be illustrated by discussing administrative controls. Although the assigned IROFS Effectiveness of Protection Indices from Table 3 above are qualitative in nature, the -2 index does correlate to nominal failure probabilities or rates for administrative controls as published in "Savannah River Site Human Error Data Base

Development for Nonreactor Nuclear Facilities."[3] The following are the recommended mean failure probabilities or rates as recommended on page 70, Table 3, of the document.

Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities	
Human Error Event – Failure of an Administrative Control	Recommended Mean Failure Probability or Rate
Nominal	5E-3
High	5E-2
Low	5E-4

The Savannah River Data Base Administrative Controls presented in the above table, however, were not bolstered by the additional scrutiny afforded by assigning them as IROFS with Management Measures (see description of Management Measures presented below after Table 5). Management Measures are required by 10 CFR 70.62(d) to ensure that IROFS are reliable and available to perform their intended functions. Thus, the failure rate data in the table do not represent failure rates for IROFS protected by Management Measures. With such measures, the failure rates would be expected to be lower. With Management Measures in place, NFS assigns a -2 to the index representing an IROFS administrative control, "protected by a trained operator performing a routine task with an approved procedure." This correlates to the information presented in Table A-10 of NUREG-1520, where an index of -2 is roughly equivalent to a probability of failure on demand of 10^{-2}. As can be seen in the Savannah River table above, that is roughly equivalent to the midpoint between the nominal and high administrative control failure probability or rate without the benefit of Management Measures assigned to the control.

The Savannah River Site Human Error Data Base Development document is applicable to NFS in that the Savannah River Site conducts operations such as storage, dissolution, and downblending of materials similar to those used for the BLEU Project at NFS. In addition, the database is particularly focused on non-reactor nuclear facilities. Further, this database was presented by the NRC, to a Facility Users Group Meeting hosted by NEI on April, 16, 2002, as a method for index standardization.

After IROFS are assigned for each credible accident scenario, Uncontrolled and Controlled Likelihoods are assessed to demonstrate the relative importance of the IROFS in preventing or mitigating the accident. The Uncontrolled Likelihood Index (T) is equal to the Initiating Event Failure Frequency Index used for the scenario. The Controlled Likelihood Index (T) is calculated by summing the Initiating Event Failure Frequency Index and the IROFS Effectiveness of Protection Index(ices). Uncontrolled and Controlled Likelihood Categories are then assigned for each scenario based on Table 4 below.

[3] Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities, (DE94012947), dated February 28, 1994.

Table 4 – Total Risk Likelihood Category	
Likelihood Category	Likelihood Index T
1	T ≤ -4
2	-4 < T ≤ -3
3	T > -3

The qualitative values for the Likelihood and Consequence Categories are cross-indexed in the Risk Matrix—Table 5 below—for categorization of Uncontrolled and Controlled Risk. The Risk (Uncontrolled and Controlled) for each accident scenario is assessed by multiplying its Consequence Category by its Likelihood Category. 10 CFR 70.61 performance requirement acceptability is determined by comparing the Controlled Risk to the values shown in Table 5. A risk greater than 4 is unacceptable and does not meet the 10 CFR 70.61 performance requirements. In that case, additional or more robust IROFS are added to meet the performance requirements, and the risk assessment process is repeated with the new IROFS. Once risk assessment has successfully been completed for each of the credible scenarios, the final set of IROFS is determined to be acceptable based on meeting the 10 CFR 70.61 performance requirements.

Table 5 – Risk Matrix			
	Likelihood Category 1 (Highly Unlikely)	Likelihood Category 2 (Unlikely)	Likelihood Category 3 (Not Unlikely)
Consequence Category 3 (High)	3 Acceptable	6 Unacceptable	9 Unacceptable
Consequence Category 2 (Intermediate)	2 Acceptable	4 Acceptable	6 Unacceptable
Consequence Category 3 (Low)	1 Acceptable	2 Acceptable	3 Acceptable

Per the requirements of 10 CFR 70.65(b)(9), NFS has defined Highly Unlikely, Unlikely, and Credible as follows.

<u>Highly Unlikely</u> – Physically possible or credible, but not expected to occur. A Credible Accident Scenario/Sequence, with a graded combination of IROFS, such as Active Engineering Controls (AEC), Passive Engineering Controls (PEC) and Administrative Controls, that mitigate or prevent the accident from occurring. It has a qualitative Likelihood Category 1, or a quantitative probability of less than or equal to 1 E-5 per accident per year. For nuclear criticality safety purposes, a system shown to provide Double Contingency protection is considered Highly Unlikely, provided that the performance requirements specified in 10 CFR 70.61 are fulfilled.

<u>Unlikely</u> – Not expected to occur during the plant lifetime. A Credible Accident Scenario/Sequence, with a graded combination of IROFS such as Active Engineering Controls (AEC), Passive Engineering Controls (PEC) and Administrative Controls, that

mitigate or prevent the accident from occurring. It has a qualitative Likelihood Category 1 or 2, or a quantitative probability of less than or equal to 1 E-4 per accident per year.

<u>**Credible**</u> – An event or accident sequence is considered 'credible' unless it is determined 'Not Credible' by meeting one of the three criteria specified below:

- An external event whose frequency of occurrence can be qualitatively estimated as having an initiating event frequency index of <u>less than or equal to</u> -5, or quantitatively determined to be <u>less than or equal to</u> 1E-6 events per year.
- A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive, excluding intent to cause harm. In order to be considered not credible, no such sequence of events can ever actually have happened in any fuel cycle facility.
- Process deviations for which there is a convincing argument, based on physical laws or engineering principles that the deviations are not possible, or extremely unlikely. The validity of the argument must not be dependent on any feature of the design or materials which is controlled by the plant's system of IROFS.

Management Measures are functions performed by the licensee, generally on a continuing basis, that are applied to IROFS to ensure the controls are available and reliable to perform their functions when needed to prevent accidents or mitigate the consequences of accidents to meet the performance requirements specified in 10 CFR 70.61. Management Measures include configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements. Management Measures are assigned to each IROFS based on its type (passive, active, etc.) and based on a graded approach to risk. Risk Reduction Level A management measures are assigned to IROFS credited with a high level of risk reduction for high or intermediate consequence events. Risk Reduction Level B management measures are assigned to IROFS credited with a moderate level of risk reduction for intermediate consequence events.

II. Discussion of Specific Chemical Accident Scenarios

In the following section we discuss the ISA's evaluation of the consequences, likelihood, and risk and the assignment of IROFS to several possible chemical accident scenarios associated with BLEU Project facilities. The purpose of the discussion is to show exactly how the ISA process worked regarding the BLEU Project and to demonstrate that the actual risk associated with the accident scenarios is even lower than what the ISA indicates.

The majority of the chemical scenarios identified in the Intervenor's Presentation fall into two categories discussed in Sections II.A and II.B below. The "Evaluations" referenced below are bounding accident sequences – accident scenarios of a similar type where all failure modes result in consequences within the same consequence category (all high, all intermediate, or all low). We also discuss these Evaluations in detail because their unmitigated consequences were assessed to be High or Intermediate and thus without safety controls they would represent the greatest potential for causing harm to people or environmental impacts. The remaining scenarios identified in the Intervenor's Presentation are process-specific and are discussed separately in Section II.C below.

A. Chemical Leaks and Spills from Tanks and/or Piping

1. Ammonium Hydroxide Leaks or Spills

Evaluations 3, 32, and 49 (from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03)

The primary chemical concerns for Evaluations 3, 32, and 49 involve the possibility of exposure to ammonia fumes due to a leak or rupture of the Ammonium Hydroxide supply header, the Bulk Ammonium Hydroxide tank, or associated supply piping for the Oxide Conversion Building (OCB)/Effluent Processing Building (EPB), which would result in a spill of ammonium hydroxide. (The Evaluations include several similar accident scenarios with similar potential consequences.) Leaks could also occur due to an excessive off-loading rate or fill rate from the recycle line when filling the bulk tank.

The use of piping and tanks to store and transport chemicals is a very common feature of chemical processing facilities. Design codes and practices have been established by nationally recognized professional organizations and are commonly utilized for NFS designs, including those associated with the BLEU Project. These codes and practices have been developed to assure safe design, therefore reducing risk to the public.

In order for the accident sequences subsumed under these Evaluations to actually occur, a combination of the following events would be needed:

1. The design would fail to consider design codes and practices to include selection of the wrong materials for the supply piping and/or tanks.
2. The installation of the supply piping would fail to follow standard installation methods.
3. Hydrotesting on the supply piping and/or tanks would not be performed properly or at all.

4. The equipment would not be properly maintained after installation.
5. Operators would have to fail to operate equipment, respond to alarms, and follow operating procedures properly when filling the bulk tank.

Because some or all of these events would have to occur to make the equipment vulnerable to leaks, spills, or ruptures, an Initiating Event Failure Frequency Index of at least -1 was assigned to the individual accident sequences included in Evaluations 3, 32, and 49.

As discussed above, because the Initiating Event Failure Frequency Index for the accident sequences in these Evaluations was at least –1, the Uncontrolled Likelihood Index was also at least –1. That translated into an Uncontrolled Likelihood Category of 3 (i.e., Not Unlikely). See Tables 4 and 5. Because these accident sequences were evaluated as having High (unmitigated) consequences, the Uncontrolled Consequence Category is 3. The Uncontrolled Risk Index then would be 3 multiplied by 3 (likelihood times consequences), or 9. Because a value of 9 in Table 5 is Unacceptable, IROFS were assigned to the accident sequences to satisfy the 10 CFR 70.61 performance requirements, i.e., to reduce the Risk Index to at least 3, which would make these accident sequences highly unlikely.

The IROFS assigned to the systems involved in the accident sequences and the protection they provide include the following:

- Correct installation of piping and tanks (includes material selection, fabrication methods, and hydrotesting) to prevent pipe or tank failure due to corrosion and/or structural failures
- Maintenance program ensures equipment is properly maintained and prevents exposure to chemical liquids or fumes due to pipe or vessel corrosion or failure
- Operating procedures and training ensure that ammonium hydroxide pump recirculation line valve remains open during pump operation to prevent potential equipment damage
- Operator response to tank high level alarm to prevent overflows
- Operating procedures and training prevent operator from allowing vendor to off load to tank unless adequate volume is available for product

These IROFS are classified as Passive Engineered Controls, Enhanced Administrative Controls, or Administrative Controls. Therefore a -2 IROFS Effectiveness of Protection Index was assigned to the majority of the IROFS. See Table 3. Due to these scenarios being identified as High consequence events, the highest level of Management Measures are applied and each accident sequence/scenario was assigned at least two IROFS, thus adding even more conservatism to the process.

The Initiating Event Failure Frequency Index for the accident sequences was at least –1 and the assignment of at least two IROFS with Effectiveness of Protection Indices totaling at least -4 to each sequence resulted in at least a -5 Controlled Likelihood Index for these scenarios. Using Tables 4 and 5, the Controlled Risk Index would then be 3. Thus, with the IROFS in place, these accident sequences are deemed to be Highly Unlikely per Table 5, and the 10 CFR 70.61 performance requirements have been met. Therefore, the risk from these accident scenarios to people and the environment is very low.

Furthermore, the risk from these potential accidents is even lower because the occurrence of any of the listed failures does not mean that catastrophic failure of the supply piping or tank and a release of a large quantity of chemicals will necessarily follow. In fact, it is very unlikely that even under failure conditions that a chemical spill large enough to affect the public or environment would occur. There are many additional factors that would have to be met and events that would have to occur. Inclusion of these factors and events in an assessment of the risk of a significant chemical exposure to the public or the environment would dramatically reduce the assessed likelihoods of these accident sequences.

First, all of the consequence evaluations were based on the conservative assumption that a leak/spill would occur when tanks and process equipment are filled to their maximum capacities. Therefore, it has been assumed that a catastrophic event occurs all the time, rather than the more likely event – a small leak that could be identified and stopped before it develops into a large leak/spill. In fact, at any given time, the contents would not be expected to be at their maximum levels. This would reduce the amount of chemical available to be spilled as a result of any accident sequence.

In addition, several IROFS would also serve to mitigate consequences. For example, operator response to a high tank level alarm would stop an overflow before the entire tanks contents are released. Maintenance programs ensure that small leaks are identified and repaired before catastrophic tank or pipe failures can occur. And finally, pump recirculation lines prevent pump damage, thereby reducing the potential for catastrophic pump failure and spills of process solutions.

Furthermore, in the event the piping or tanks were to begin leaking, additional controls are in place to prevent a spill from developing into a large event. These controls include a dike containment area for the bulk supply tank and a Spill Response Plan that provides instructions for cleaning up chemical spills. BLEU Complex facility work areas are staffed with employees 24 hours a day, 7 days a week. Therefore, even if a spill were to occur, staff would identify the event and enact the Spill Response Plan long before the public or the environment could be affected at the levels identified in the ISA Summary. These mitigating factors were conservatively not included in the risk assessment for these accident sequences in the ISA.

Second, a number of conservative factors in the consequence assessments make it unlikely that maximum amounts of ammonia vapors would be dispersed to the environment or that maximum exposures to individuals would result, even if a large spill were to occur. All accident sequence evaluations that resulted in high consequences were based on conservative evaporation rates at a maximum outdoor temperature of 90° F, using worst-case wind speed and direction and atmospheric stability for dispersion. Lower temperatures that would be characteristic of the Erwin, Tennessee location most of the time would result in lower evaporation rates and therefore lower concentrations at off-site locations. A greater wind speed or lower atmospheric stability would result in greater dispersion and thus lower individual exposure. It was also assumed that exposed individuals were at the closest site boundary and that the wind was blowing in their direction. Each of these assumptions leads to an overestimate of exposures. The intent of the exposure calculations performed for the ISAs is to present conservative exposure values. Use of more realistic assumptions would significantly reduce the estimated consequences for the chemical accident scenarios. See Declaration of John R. Frazier Regarding the Dispersion of Airborne Effluents (Dec. 14, 2004).

2. Other Evaluations Involving Chemical Leaks or Spills

There are a number of other BLEU Project accident sequence evaluations involving chemical leaks and spills that are similar in nature to the ammonia spills discussed above. These include:

- Nitric acid spills that occur outside (Evaluation 3 from Table 4-5 of the ISA Summary for UAL and Downblending, Rev. 0, 10/11/02, Evaluation 3 from Table 4-5 of the ISA Summary for BPF, Rev. 1, 2/6/04, and Evaluations 30, 47, and 48 from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03),
- Nitric acid spills that occur inside (Evaluation 4.1 from Table 4-5 of the ISA Summary for BPF, Rev. 1, 2/6/04),
- Deionized water overflow of tanks in Downblending Area that occur outside (Evaluation 29 from Table 4-5 of the ISA Summary for UAL and Downblending, Rev. 0, 10/11/02, and Evaluation 29.1 from Table 4-5 of the ISA Summary for BPF, Rev. 1, 2/6/04),
- Liquid waste spills that occur outside (Evaluation 57 from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03),
- Hydrogen peroxide spill that occurs outside (Evaluation 22 from Table 4-5 of the ISA Summary for UAL and Downblending, Rev. 0, 10/11/02, and Evaluation 22 from Table 4-5 of the ISA Summary for BPF, Rev. 1, 2/6/04), and
- Caustic tanker spill that occurs outside (Evaluation 35 from Table 4-5 of the ISA Summary for BPF, Rev. 1, 2/6/04).

All of these spills require the same or similar combination of events to occur, have the same or less frequent likelihood indices, and the same or similar IROFS assigned to them. In addition, for all of the spills that occur outdoors, the same conservative assumptions apply to the consequence evaluations. For the one inside spill, i.e., spill of nitric acid that is transferred from the bulk chemical tank, most of the same conservative assumptions also apply (maximum tank contents, no operator action to shut down flow, conservative atmospheric dispersions assumptions).

In conclusion, similar to the ammonia accidents discussed in detail above, several low probability events that were conservatively not included in the risk assessment for these accident sequences/scenarios would also have to occur simultaneously and several mitigating factors that were conservatively not included in the consequence assessments would have to fail to take effect before a chemical spill that would significantly affect the public or the environment would occur as a result of these accident scenarios. These conservative assumptions include:

- Any leak/spill would occur when tanks and process equipment are filled to their maximum capacities
- Spill containment areas and Spill Response Plan not taken into account
- Conservative atmospheric dispersion assumptions (evaporation rate, temperature, wind stability and direction)
- Leak and spill amounts were assumed not to be mitigated by IROFS to less than maximum process equipment capacities

B. Ammonia Vapor Release Due to Fire

Evaluations 38, 39, and 40 (from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03)

The primary chemical concerns for Evaluations 38, 39, and 40 involve the possibility of exposure to ammonia vapors due to a fire in the OCB Tank Gallery, a fire in the EPB, or a fire on the second floor of the OCB resulting in a release of ammonia vapors from tanks or equipment.

In order for these accident sequences to occur, some combination of the following events would be necessary.

1. Employees would have to bring combustible materials into areas where they are strictly prohibited.
2. A fire initiator would have to be in the presence of combustible material long enough to start a fire.
3. A fire would have to burn unnoticed long enough to move into the areas of concern and affect the tanks or equipment such that ammonia vapors could escape.
4. The fire suppression/detection systems would have to fail to activate thus allowing the event to continue indefinitely, potentially breaching the building and allowing the vapors to escape to the environment.

Because some or all of these events would have to occur to make the equipment vulnerable to release vapors if a fire occurred, an Initiating Event Failure Frequency index of -1 was assigned to the individual accident sequences included in Evaluations 38, 39, and 40.

As similarly discussed above with respect to the ISA process generally and the ammonium hydroxide accident sequences, because the Initiating Event Failure Frequency Index for the accident sequences in these Evaluations was –1, the Uncontrolled Likelihood Index was also –1. Thus, the Uncontrolled Likelihood Category was 3 (i.e., Not Unlikely). See Tables 4 and 5. Because these accident sequences were evaluated as having High (unmitigated) consequences, the Uncontrolled Consequence Category is 3. The Uncontrolled Risk Index then would be 3 multiplied by 3 (likelihood times consequences), or 9. Because a value of 9 in Table 5 is Unacceptable, IROFS were assigned to the accident sequences to satisfy 10 CFR 70.61

performance requirements, i.e., to reduce the Risk Index to at least 3, which would make these accident sequences highly unlikely.

The IROFS assigned to the systems involved in the accident sequences and the protection they provide include the following:

- Combustible loading program restricting the amount of potentially combustible material in the operating spaces of the facilities
- Fire protection test, maintenance and inspection activities detect and remove potential combustibles from the operating spaces of the facilities

Both of these IROFS are classified as Administrative Controls, therefore a -2 IROFS Effectiveness of Protection Index was assigned to each. Due to these scenarios being identified as High consequence events, the highest level of Management Measures are applied.

The Initiating Event Failure Frequency Index for the accident sequences was –1 and the assignment of at least two IROFS with Effectiveness of Protection Indices totaling -4 to each sequence resulted in a -5 Controlled Likelihood Index for these scenarios. Using Tables 4 and 5, the Controlled Risk Index would then be 3. Thus, with the IROFS in place, these accident sequences are deemed to be Highly Unlikely per Table 5, and the 10 CFR 70.61 performance requirements have been met. Therefore, the risk from these accident scenarios to people and the environment is very low.

Additional mitigating factors in place that were not included in the risk assessment include Fire suppression/Automatic Sprinkler Systems for the Oxide Conversion and Effluent Processing Buildings and a Fire detection system for the second floor of the Oxide Conversion Building.

Furthermore, the risk from these potential accidents is even lower because the occurrence of any of the listed failures does not mean that the maximum release of ammonia vapors to the environment will follow. In fact, it is very unlikely that even if these accident sequences occur that a release of ammonia vapors large enough to affect the public or environment would occur. There are many additional factors that would have to be met and events that would have to occur. Inclusion of these factors and events in an assessment of the risk of a significant chemical exposure to the public or the environment would dramatically reduce the assessed likelihoods of these accident sequences.

A number of factors conservatively not included in the consequence assessments would preclude the dispersion of maximum amounts of ammonia vapors to the environment. First, all of the consequence evaluations were conservatively based on a fire occurring when tanks and process equipment are filled to their maximum capacities. In fact, at any given time, the contents would not be expected to be at their maximum levels. This would reduce the amount of chemical available to be spilled as a result of any accident sequence.

Second, if the fire were small and contained, as would be the case when the fire suppression systems function correctly, then the ammonia vapors would be released from the elevated stack, resulting in significant dispersion and reduced concentrations to which affected members of the public would be exposed. If the fire remained uncontained, resulting in breaches of the roof, then the additional plume rise generated by a high-temperature fire would also result in reduced concentrations to which the environment and the public would be exposed. Therefore, the assumption that a catastrophic event occurs all the time, rather than the more likely event – a small fire that could be identified and stopped before it develops into a large fire with significant consequences is indeed conservative.

In addition to the foregoing conservative assumptions, the consequence assessments conservatively did not account for the mitigating effects of the NFS Emergency Response Plan. BLEU Complex facility work areas are staffed with employees 24 hours a day, 7 days a week. If a fire was discovered that was large enough to overwhelm the fire suppression/detection systems, staff would identify the event and enact the Emergency Response Plan long before the public or the environment could be affected at the levels identified in the ISA Summary. The Emergency Response Plan includes instructions for actions to be taken during different types of emergencies. Those actions may include activating local emergency response groups to respond to a fire.

Finally, as with the other chemical accident sequences, consequence assessments were based on conservative assumptions regarding outdoor temperatures, wind speed and direction, and atmospheric stability, which pertain to chemical dispersion and potential off-site exposure of individuals. These assumptions result in conservative exposure values. Use of more realistic assumptions would significantly reduce the estimated consequences for the chemical accident scenarios.

In conclusion, as with the other chemical accident scenarios discussed previously, multiple low probability events that were not included in the risk assessment for these accident scenarios would also have to occur simultaneously before a release of ammonia vapors large enough to affect the public or the environment would be possible as a result of these accident scenarios. Moreover, the consequences of any fire followed by the release of ammonia vapors would very likely be much less than those described in the consequence assessments because of the significant mitigating factors that were not considered in that assessment. Thus, in the end, the risk of significant environmental impacts resulting from these accident sequences is extremely low.

C. Process-Specific Scenarios

The remaining BLEU Project accident sequence evaluations identified in the Intervenor's Presentation require specific process upsets to occur as initiating events to set each accident scenario in motion. All scenarios would occur inside facility buildings. These include:

- Glovebox enclosure explosion (Evaluation 21 from Table 4-5 of the ISA Summary for UAL and Downblending, Rev. 0, 10/11/02, and Evaluation 27.1 from Table 4-5 of the ISA Summary for BPF, Rev. 1, 2/6/04);
- Leak from ammonia recovery equipment (Evaluation 27 from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03);

- Explosion of ammonium nitrate solution in systems containing >1% weight (Evaluation 33 from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03);
- NOx release due to addition of a drum of enriched scrap material to the natural dissolver (Evaluation 55 from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03);
- Release of calciner off-gas resulting in release of ammonia and/or hydrogen to the room (Evaluations 23 and 59 from Table 4-4 of the OCB/EPB ISA Summary, Rev. 0, 11/14/03);

The ISA process used to evaluate the consequences of these scenarios and assess their likelihood and risk followed the procedures outlined above.

All of these scenarios require specific process upsets to occur as initiating events, have similar likelihood indices as those discussed previously in Section II, and have multiple process-specific IROFS assigned to them. In addition to the IROFS assigned to each scenario in these evaluations, there are many other pieces of equipment that would indicate changes in process parameters thus alerting the operator if the process was not operating properly. In many cases, the operator(s) would know almost immediately if the process was not operating properly due to equipment being connected through a Central Control System – any part of the system can be monitored by any worker at any work station computer. These indicators would allow the operator(s) to intervene and prevent or mitigate the consequences of the accident scenarios. Thus, even more conservatism is provided here than in the more general scenarios discussed above.

In conclusion, similar to the accidents discussed in Section II.A and II.B in detail above, multiple low probability events that were conservatively not included in the risk assessments for these accident sequences/scenarios would also have to occur simultaneously and several mitigating factors that were conservatively not included in the consequence assessments would have to fail to take effect before a chemical spill or release that would significantly affect the public or the environment would occur as a result of these accident scenarios. These conservative assumptions include:

- Any leak/spill/release would occur when process equipment is filled to maximum capacities.
- Spill Response Plan actions were not taken into account.
- Building process ventilation was assumed to not be available.
- Leaks/spills/releases were assumed not to be mitigated by IROFS to less than maximum process equipment capacities.
- For the glovebox enclosure or calciner off-gas hydrogen explosions, any excess hydrogen is conservatively assumed to result in a maximum explosion. In reality, small amounts of excess hydrogen are removed by the process ventilation system and never result in explosions.

- For the ammonium nitrate explosion, any concentration greater than 1% was assumed to result in explosion when, in fact, only concentrations greater than 92% have been shown to result in detonations.
- IROFS such as hydrogen gas analyzers and purge valves, high temperature and pressure indicators and interlocks, and enrichment monitors, which also serve to limit accumulations of potentially explosive materials to less than explosive levels, are assumed not to be available.

The result of all of these conservatisms is that the risk from process-specific chemical accident scenarios is significantly less than what is indicated by the ISA alone.

III. Conclusion With Respect to Chemical Accident Scenarios

In conclusion, the chemical scenarios discussed in Section II require multiple low probability events that were not included in the risk assessment to occur simultaneously before a chemical release large enough to affect the public or the environment would be possible as a result of these accident scenarios. Moreover, the consequences of any of these scenarios followed by a chemical release would very likely be much less than those described in the consequence assessment because of the conservative assumptions that were made and the significant mitigating factors that were not considered in that assessment. Thus, in the end, the risk of harm to people or significant environmental impacts resulting from these accident sequences is extremely low.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 15, 2004.

Jennifer K. Wheeler

Jennifer K. Wheeler

Carol L. Mason

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I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 15, 2004.

Jennifer K. Wheeler

Carol L. Mason

Carol L. Mason

Statement of Qualifications

Jennifer K. Wheeler

I have been in my current position as Integrated Safety Analysis (ISA) Manager for Nuclear Fuel Services, Inc. since July 2002. My academic degrees include a B.S. in Civil Engineering and an M.S. in Civil Engineering. I have over 12 years of professional experience in civil engineering, primarily in the areas of storm water system and roadway design and construction. I was licensed by the Tennessee State Board of Architectural and Engineering Examiners to practice engineering in 1997, and my registration has been renewed every two years since that date, with the most recent renewal valid through 2005. I am a member of the National Society of Professional Engineers and the American Society of Civil Engineers. My further qualifications and experience as an engineer and manager are detailed in my resume (attached).

I am familiar with the terms and concepts related to ISA due to my position as Integrated Safety Analysis Manager for NFS. In 2002, I participated in ISA Leader Training led for NFS by The Process Safety Institute of ABS Consulting, Knoxville, TN. I have been involved in many discussions with the NRC regarding ISA, and many of those discussions related to the ISAs for the BLEU Project. I have personally supervised the compilation of consequence analysis results and risk assessment for twelve (12) ISA Summaries, two (2) of which related to the BLEU Project. I have personally contributed to or reviewed four (4) additional ISA Summaries, all of which related to the BLEU Project. I am familiar with the concepts and procedures for developing Process Hazard Analyses, and this allows me to understand and execute the process for identifying accident scenarios/sequences. Through the supervision of safety analysts, I am familiar with the concepts, models, and procedures for assessing chemical consequences. I am familiar with the chemical processes associated with the BLEU Project through review of Piping & Instrumentation Drawings, interactions with Operations personnel, visits to a similar facility in Richland, Washington, and knowledge of similar processes already in operation at NFS.

JENNIFER K. WHEELER, P.E., *Integrated Safety Analysis Manager*, Nuclear Fuel Services, Inc.

Education:

Master of Science	Civil Engineering, University of Tennessee, 2000
Bachelor of Science	Civil Engineering, Clemson University, 1990

Experience Summary:

Ms. Wheeler's current assignment is Integrated Safety Analysis Manager at NFS. In this capacity, Ms. Wheeler manages the Integrated Safety Analysis (ISA) program and supervises preparation of ISA Summaries. The purpose of the ISA program is to identify potential accidents at NFS, including chemical accidents, and designate the items relied on for safety necessary to prevent those potential accidents and/or mitigate their consequences. Through the supervision of safety analysts, Ms. Wheeler is familiar with the concepts, models, and procedures for assessing chemical consequences. Ms. Wheeler's duties also include: ensuring compliance with regulatory requirements and guidance documents, maintaining and updating the ISA and supporting ISA documentation, leading or participating in the process to evaluate, implement, and track changes to the NFS site, processes, equipment, structures, and personnel activities. Ms. Wheeler has also lead or participated in negotiations with regulatory agencies regarding ISA related matters for three NRC License Amendments regarding the BLEU Project – the Uranyl Nitrate Building, the BLEU Preparation Facility, and the Oxide Conversion/Effluent Processing Buildings.

Employment History:

Nuclear Fuel Services, Inc., Erwin, TN – *Integrated Safety Analysis Manager*, 2002 – present
Nuclear Fuel Services, Inc., Erwin, TN – *Project Manager*, 2001 – 2002
Nuclear Fuel Services, Inc., Erwin, TN – *Process Engineer III*, 2000 – 2001
City of Johnson City, Johnson City, TN – *Civil Engineer III*, 1996 – 2000
City of Johnson City, Johnson City, TN – *Development Specialist*, 1995
Virginia Dept. of Transportation, Richmond, VA – *Transportation Engineer*, 1994 – 1995
Virginia Dept. of Transportation, Richmond, VA – *Transportation Engineer Trainee*, 1992 – 1994
Virginia Dept. of Transportation, Fairfax, VA – *Transportation Engineering Technician*, 1991 – 1992
Virginia Dept. of Transportation, Fairfax, VA – *Highway Construction Inspector Trainee*, 1990 – 1991

Professional Development and Achievements:

- Registration by the Tennessee State Board of Architectural and Engineering Examiners, *Professional Engineer*, 1997 – present, Certificate Number 103481
- National Society of Professional Engineers, 1997 – present
- American Society of Civil Engineers, 1996 – present
- ISA Leader Training, The Process Safety Institute of ABS Consulting, Knoxville, TN, 2002

NFS ISA Publications (prepared and submitted to support NRC License Amendments under the direction of or contributions made by Ms. Wheeler):

- *Integrated Safety Analysis Summary for the Blended Low-Enriched Uranium (BLEU) Project – Uranyl Nitrate Building (UNB), Rev. 1.* August 2002.
- *Integrated Safety Analysis Summary for the Blended Low-Enriched Uranium Preparation Facility, Rev. 0.* October 2002.
- *Integrated Safety Analysis Summary for the Blended Low-Enriched Uranium (BLEU) Project – Uranyl Nitrate Building (UNB), Rev. 2.* May 2003.
- *Integrated Safety Analysis Summary for the Blended Low-Enriched Uranium (BLEU) Project – Oxide Conversion and Effluent Processing Buildings, Rev. 0.* October 2003.
- *Integrated Safety Analysis Summary for the Blended Low-Enriched Uranium Preparation Facility, Rev. 1.* February 2004.
- *Integrated Safety Analysis Summary for the Blended Low-Enriched Uranium (BLEU) Project – Oxide Conversion and Effluent Processing Buildings, Rev. 1.* August 2004.

NFS ISA Publications (prepared and submitted to support 10 CFR 70.62(c)(3)(ii) under the direction of and contributions made by Ms. Wheeler):

NFS Site ISA Summary, Rev. 0. October 2004.
300 Complex Production Areas 100 to 900 Integrated Safety Analysis Summary, Rev. 0. October 2004.
300 Complex Recovery Integrated Safety Analysis Summary, Rev. 0. October 2004.
300 Complex Support Systems Integrated Safety Analysis Summary, Rev. 0. October 2004.
Building 105 Laboratory Integrated Safety Analysis Summary, Rev. 0. October 2004.
Building 310 Warehouse Integrated Safety Analysis Summary, Rev. 0. October 2004.
Building 300 Warehouse Integrated Safety Analysis Summary, Rev. 0. October 2004.
Building 100 NDA Laboratory Integrated Safety Analysis Summary, Rev. 0. October 2004.
Laboratory 110B, 110D, and 131 Integrated Safety Analysis Summary, Rev. 0. October 2004.
Waste Water Treatment Facility Integrated Safety Analysis Summary, Rev. 0. October 2004.

CAROL L. MASON

Education:

M.S., Chemical Engineering, 1975 (University of Tennessee)
B.S., Chemical Engineering, 1970 (University of Tennessee)

Security Clearance: DOE Q

Experience: Ms. Mason has over 25 years of experience in safety analysis, reliability analysis, and risk assessment for nuclear and non-nuclear facilities for Nuclear Fuel Services (NFS), and at DOE's Miamisburg Cleanup Project (MCP), Savannah River Site (SRS), Oak Ridge National Security Complex, Oak Ridge National Laboratory (ORNL), East Tennessee Technology Park (ETTP), Hanford, Rocky Flats Environmental Technology Site (RFETS), and other DOE sites, as well as the Paducah and Portsmouth Gaseous Diffusion Plants. Ms. Mason's safety analysis expertise includes identification of accident initiators, accident sequence development, source term analysis, frequency quantification, and application of atmospheric dose calculation codes to estimate on-site and off-site consequences. She has developed and implemented methodologies and databases for conducting chemical hazard analysis for hazards screening and accident analysis. Ms. Mason is an experienced Technical Task Leader in all areas related to safety and risk analysis and complete safety authorization-basis documentation.

Employment History

Senior Engineer, Science Applications International Corporation (SAIC), Oak Ridge, TN, 1977 to present.

Ms. Mason's current assignment involves chemical hazard and accident analysis for fuel production and support facilities at the NFS site in Erwin, TN. Her responsibilities include quantification of source terms and consequences, risk assessment, and complete integrated safety analysis documentation (2003 to present).

Ms. Mason's primary assignment for the past five years has involved developing safety authorization-basis documentation for nuclear and non-nuclear facilities and operations for CH2MHill, Inc. at MCP. Her responsibilities for these projects include identification of accident initiators, accident sequence development, quantification of source terms and consequences, and complete authorization-basis documentation. She has also provided technical review and has contributed specific technical expertise in the areas of chemical hazard analysis and application of atmospheric dose calculation codes to more than ten other MCP projects. Based on her experience with, and understanding of, all safety authorization-basis documents, she was responsible for four annual updates of the emergency management hazard assessments for the entire MEMP site (1997 to present).

Ms. Mason was the Task Leader for the documented safety analysis (DSA) for the Mars Exploration Rover (MER) ground operations involving light-weight radioisotope heater units. The MER Project supports NASA's Office of Space Science's plan for the exploration of the solar system. The U.S. DOE supports this mission by providing the radioisotope heater units that proving heating of the rover electronics on the surface of Mars. Because DOE supplies the heater units, all ground operations involving the rover are considered a "DOE Nuclear Activity" and require a DSA under the definition of 10 CFR 830, Subpart B, for the time period when the heater units are present. Based on previous experience with safety analysis for assembly of these units, Ms. Mason was asked to lead the team that developed the DSA for the final steps: receipt and storage of the units at Kennedy Space Center (KSC), continuing through payload and spacecraft integration, spacecraft to third-stage mating, and conclude with attaching the spacecraft/third-stage assembly to the top of the Delta II launch vehicle at Space Launch Complex 17 on the eastern range of Cape Canaveral Air Force Station (CCAFS). Because 10 CFR 830 is relatively new (complete DOE-wide compliance is required by April 2003), this is the first time that a DSA has been developed for ground operations involving DOE-supplied materials at facilities located at other than DOE sites (2002 to present).

Assembly of radioisotope heater units for use in NASA space missions has recently been relocated to Argonne National Laboratory-West (ANL-W). Ms. Mason led the preliminary DSA (PDSA) effort for modifications to the facility at ANL-W where this project will be located. This PDSA was developed in parallel with the Title I and Title II design efforts, and was also be a 10 CFR 830-compliant PDSA.

Ms. Mason has provided hazard and accident analysis support for several environmental assessments (EAs) and environmental impact statements (EISs) including the Mercury Management EIS and Mercury Reflasking EA for the Defense National Stockpile Center (DNSC); and Programmatic EA for Management of Potentially Reuseable Uranium, EA for the Receipt and Storage of Uranium Materials from the Fernald Environmental Management Project Site, and the Programmatic EA for Treatment and Disposal of Low-Level Mixed Waste for the DOE.

Ms. Mason evaluated off-normal operating conditions for the Paducah and Portsmouth Gaseous Diffusion Plants for the SAR upgrade program. She has also served as Task Leader for graded approach SARs for waste storage facilities at the Y-12 Plant and ORNL, and developing BIOs and bases for hazard categorization for uranium processing buildings at ETTP (1993 to 1997).

Ms. Mason has served since 1985 as task leader and principal contributor for system and safety analyses of SRS facilities and processes. Major projects include F- and H-Canyons, A-Line, FB-Line, Production Control Facilities, Effluent Treatment Facility, Transuranic Waste Facility, In-Tank Precipitation Process, Defense Waste Processing Facility, Building 235-F, and Savannah River Technology Center (SRTC). Her responsibilities for these projects included identification of accident initiators, accident sequence development, frequency quantification, quantification of source terms, and evaluation of safety-related systems. She also provided technical review for special studies and has contributed specific technical expertise in the areas of unreviewed safety questions, double contingency analysis, chemical hazards analysis, and application of atmospheric dose calculation codes to more than a dozen other SRS projects (1993 to 1996).

Ms. Mason participated in DOE-HQ safety surveys at the Y-12 Plant (1991 to 1992). Bounding offsite exposure accidents were identified based on review of existing safety documentation, facility walkthrough, and familiarization with operating practices. Atmospheric source terms were determined using standardized methods developed for use across the weapons complex.

Ms. Mason developed and quantified fault trees, described accident sequences, and quantified source terms as part of a bounding safety assessment for decontamination and decommissioning (D&D) activities at RFETS (1989 to 1990). D&D activities covered by the assessment included waste packaging and storage, onsite waste transportation, tank draining, and pipe cutting and capping.

Ms. Mason provided support to the Y-12 Safety Review and Documentation Program. She developed facility and process descriptions as well as criticality fault trees for the O-Wing SAR. She also performed accident analysis for the Powder Production Prototype for the Lithium Process Replacement (LPR) project, and reviewed safety documents for other processes and facilities associated with the LPR (1987-1990).

Ms. Mason helped conduct a Level 1 probabilistic risk assessment (PRA) of the SRS production reactors in 1988. She analyzed component failure data, implemented human reliability analysis (HRA) into the fault tree databases, and developed input files for component fault trees.

Ms. Mason developed complete fault tree analyses for several safety-related and major support systems for the Level 1 PRA of the N-Reactor at Hanford (1986 to 1987). She established system interfaces and ensured that all fault tree interface and support system dependencies were properly modeled. Ms. Mason also supported a limited-scope PRA for the N-Reactor, which included assessing the confinement response to both internally and externally initiated accident sequences, developing and quantifying accident sequences, and quantifying system fault trees.

Ms. Mason's nuclear power plant experience includes evaluating pressurized thermal shock (PTS) at nuclear power plants, developing RELAP5 and RETRAN primary system models as part of the PTS program, and developing system level FMEAs for various NPP systems. She also developed a categorization scheme for NPP systems and components, and provided data analysis and validation for the In-Plant Reliability Data Base for ORNL based on this categorization. She contributed to the development of the reliability data base for the Swedish State Power Board Ringhals 2 risk assessment (1980 to 1984).

Ms. Mason developed modularized probabilistic reliability/availability models for a systems engineering evaluation of DOE's Strategic Petroleum Reserve during 1982 and 1983. She developed system and site models and methodology to estimate availability for both nominal and degraded state configurations for all phases of various operating modes. This methodology was used to combine availability estimates with event tree flow rates to yield site availability predictions. Ms. Mason also developed reliability block diagrams for all operating phases and modes.

From 1977 to 1980 Ms. Mason helped conduct a risk assessment to support SAR preparation, a preliminary hazards analysis, and several systems analyses for the Centrifuge Plant Demonstration Facility at the K-25 Site (now ETTP). She performed reliability and availability analyses, prepared system design descriptions for specific systems for the centrifuge cascade, and prepared acceptance and test plans.

Ms. Mason also provided support to the Energy Division at ORNL. She developed an assessment of the capability and availability of state-of-the-art pollution control equipment as part of a technology assessment for an Atmospheric Fluidized Bed Combustor (AFBC) Demonstration Plant, provided engineering and economic assessment of alternative processing and energy production technologies in the pulp and paper industry, and assessed the technical feasibility of converting oil- and gas-fired boilers to coal (1977 to 1978).

Research Fellow, University of Tennessee, 1974 to 1977.

As a Research Fellow, Ms. Mason developed an experimental program designed to identify sulfur-sorbent materials for use in fluidized bed coal combustion.

Associate Development Engineer, UCC-ND Oak Ridge Gaseous Diffusion, 1970 to 1974.

Ms. Mason was an Associate Development Engineer in charge of program analysis with responsibility for design of experimental and quality control testing programs.

Affiliations:

American Institute of Chemical Engineers

ATTACHMENT 2

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Presiding Officer

In the Matter of	)	
	)	Docket No. 70-143
Nuclear Fuel Services, Inc.	)	Special Nuclear Material
	)	License No. SNM-124
(Blended Low Enriched Uranium Project)	)	

Declaration of Robert L. Frost Regarding NFS Response to Criticality Accident Sequences Cited by Intervenors in Their Written Presentation

Robert L. Frost states as follows under penalty of perjury:

Intervenors Sierra Club et al. cited a number of criticality accident sequences from the BLEU Project Integrated Safety Analyses (ISAs) in their written presentation (brief) as evidence to support their claims regarding accident risk associated with the BLEU Project. The accident sequences they quoted all had Controlled Likelihood Indices of -4 or -5. Intervenors have mistakenly asserted that those accident sequences have probabilities of 10^{-4} and 10^{-5} per year, respectively; i.e., they interpret a likelihood index of -4 to indicate that the probability of an accident occurring is 10^{-4} per year. However, intervenors have neglected the facts that (1) the likelihood indices are conservative estimates and (2) once the ISAs demonstrate that the criticality accident sequences are highly unlikely the analysis stops and they do not go on to assess the actual probability of each sequence. Thus, intervenors have mistakenly concluded that the probabilities of potential criticality accidents associated with the BLEU Project are significantly greater than they actually are.

The Controlled Likelihood Index is a qualitative indication of the likelihood of an accident sequence that is used to demonstrate, via the ISA, compliance with the requirements of 10 CFR § 70.61. While the indices correspond with approximate orders of magnitude of probability, conservatively estimated, they are presented qualitatively as permitted by the regulations. Section 70.61 requires accidents with "high" consequences to be at most highly unlikely and accidents with "intermediate" consequences to be at most unlikely. Analysis of the accident sequences is performed in a manner to demonstrate that this license criterion is met.[1] Thus, the accident sequence analysis is not a strict probabilistic analysis of the expected frequency of occurrence or probability of that sequence.

[1] The ISA process is discussed in greater detail in the Declaration of Jennifer K. Wheeler and Carol L. Mason Regarding NFS Response to Chemical Accident Sequences Cited by Intervenors in Their Written Presentation.

In addition, according to 10 CFR 70.61, a license applicant only has to show that potential criticality accidents (whose consequences are presumed by regulation to be high) are highly unlikely. There is no requirement that the ISA go further to demonstrate their actual probabilities or to show that they are not credible or how far below the point of credibility their likelihoods really are. Once it is shown that an accident sequence is highly unlikely, the ISA need not address any additional unlikely events that would have to occur before the accident sequence would occur and it need not address the effects of any additional safety systems that would in fact make the sequences less likely. As discussed below, all of the accident sequences cited by intervenors require additional unlikely events to occur and/or additional safety systems to fail before the sequences would occur. Thus, the actual likelihoods of the accident sequences are significantly lower than what is indicated by the ISA Controlled Likelihood Indices alone.

The following sections discuss each of the accident scenarios or sequences that the intervenors referenced in their presentation. All of them had -4 or –5 Controlled Likelihood Indices in the ISA. The Controlled Likelihood Index listed in the ISA is based on the Effectiveness of Protection Index for the controls that prevent the accident from occurring. These controls are referred to as IROFS (Items Relied On For Safety). In a limited number of cases credit is also taken for the anticipated likelihood of occurrence of an initiating or enabling event or series of events. The purpose of the discussions that follow is to demonstrate the inherent conservative nature of these evaluations, and to point out all the things that would have occur beyond those credited in the ISA before a criticality would be possible. By doing so we show that the intervenors have significantly overestimated the probability of criticality accidents associated with the BLEU Project.

The discussions that follow demonstrate that there is a very low probability that a criticality accident would occur in the NFS BLEU facilities. Regarding potential consequences, a criticality accident in the BLEU facility would most likely have "high" consequences (as defined in 10 CFR 70.61) for any unfortunate onsite workers who were close by when the event occurred. Off-site consequences to members of the public and the environment, though, would almost certainly be low. This is evidenced by the dose data from the criticality accident at Tokai-Mura, Japan. In the highly unlikely event a criticality accident were to occur at the BLEU facility, consequences to off-site members of the public and the environment would be less than those from the Tokai-Mura accident due to the existence of the detailed NFS emergency plan that includes provisions for bringing the accident under control as well as coordination with local emergency response organizations. The potential consequences of a criticality event are addressed further in Declaration of Robert L. Frost and John R. Frazier Regarding Intervenor's Claims of Consequences From the Tokai-Mura, Japan Criticality Accident.

I. Container Spacing Violations in the Uranium Metal Dissolution Area

Intervenors' brief cites two criticality accident sequences involving the mishandling of containers of HEU. Intervenors' Pres. at 29-30. They cite the following accident sequences in the BLEU Preparation Facility (BPF) U Metal Dissolution Process Area:

4.1.26.4.1	Container spacing upset with process equipment with only one operator handling portable containers (ISA likelihood index of -4)
4.1.26.4.1.b	Container spacing upset with process equipment with only one operator handling portable containers (ISA likelihood index of -4)
4.1.26.4.2	Container spacing upset with storage racks with two or more operators handling portable containers (ISA likelihood index of -4)

The use of small bottles and cans to store and transport HEU is a very common feature of all facilities that process this material. Safe procedures for storage and transport of these containers have been established and are commonly observed throughout all NFS operations, including those associated with the BLEU Project. These procedures are designed to assure a minimum separation distance is maintained between containers and between containers and other equipment that may contain HEU.

The particular containers referred to in these accident sequences are steel cans, with an approximate diameter of 5 inches and an approximate height of 10 inches. Each can contains uranium metal, either as a single piece or as several pieces. These cans are shipped to NFS, unloaded in the receiving area, and stored in the main vault. They are brought into the BPF uranium dissolution area on an as needed basis, and loaded into a glovebox. Inside the glovebox the cans are opened, and the uranium metal is removed and placed in a dissolver. The empty cans are removed from the glovebox and disposed of.

The three accident sequences listed above all involve violations of the controls (IROFS) that assure containers (in this case the metal cans) remain properly spaced. Those controls are:

1. Only one container may be hand carried per person at a time

2. Hand carried containers must be spaced at least 12 inches from each other and from process equipment

Several different accident sequences could result from failure of these two controls. The examples discussed here involve a single person; the multi-person scenarios are very similar. A criticality is theoretically possible if an operator were to hand carry <u>two or more</u> containers simultaneously, in violation of requirement number 1 above, <u>and</u> then place his hand carried containers in contact with each other <u>and</u> with a piece of equipment containing HEU. By contrast, if the person were to hand carry a single container, and, in violation of requirement number 2 above, place that container in contact with one other container or with equipment containing HEU, <u>no</u> criticality would occur. It is instructive to look at this scenario more closely.

First, note that in order for this accident sequence to proceed, the operator must make multiple violations of criticality safety requirements:

1. He first must pick up and carry more than one container, in violation of requirement (1) above.

2. He then must ignore the 12-inch spacing requirement for those containers, in violation of requirement (2) above.

3. Finally, he must ignore the 12-inch spacing requirement between containers and process equipment, again in violation of requirement (2) above.

Each of the steps listed above is a violation of an IROFS. Each of these administrative controls are routinely observed by operators in their daily operations, and are backed up by extensive operator training. As discussed below, the controls also incorporate a large margin of safety. This makes assignment of a -2 Effectiveness of Protection Index for each of these IROFS appropriate. Summing the failure frequencies leads to a -4 Controlled Likelihood Index for the accident sequence. The -4 index corresponds to a determination that the accident sequence is highly unlikely and meets the requirements of 10 C.F.R. § 70.61.

However, the fact that these violations occur does not assure a criticality accident will follow. In fact, it is very unlikely even under these conditions that a criticality would occur. There are many additional factors that would have to be met. Consideration of these factors shows that the actual likelihood that this accident sequence would lead to a criticality accident is significantly lower than even what the ISA analysis indicates.

First, safety control (IROFS) number 2 is that a 12-inch separation must be maintained between containers of HEU, or between such containers and process equipment that contains HEU. Violation of this requirement is conservatively assumed to result in no separation between the containers/equipment. This is a crucial point. The requirement is at least 12 inches of separation. Therefore, having 11 inches of separation is a violation of the requirement – but it is <u>not</u> sufficient to cause a criticality accident. In fact, the separation must be less than half an inch for criticality to be possible. Consider what this means: a person has to hold two containers of HEU within ½ inch of each other, and then put those two also within ½ inch of process equipment containing HEU. There is absolutely no motivation for an operator to do such a thing. In an accident scenario it is much less likely that the containers and equipment would be accidentally placed within half an inch of each other than within 12 inches of each other. Nevertheless, in the ISA process a -2 Effectiveness of Protection Index was conservatively assigned to the failure to maintain the required 12 inch spacing; there was no distinction between a small violation (11 inch spacing) and a significant one (1/2 inch or less spacing).

Second, it is also important to understand that the ISA conservatively assumed that the containers of HEU and the equipment <u>all</u> contained a significant amount of material when in fact this is often not the case. The containers of HEU have differing amounts and types of HEU in them. The mass of uranium metal in the cans shipped to NFS varies, with a maximum value of 11 kg and an average value of 9 kg. Of the thousands of cans of uranium metal used at NFS, less than 10 contain this maximum loading. So the likelihood that the two cans involved in an accident would both contain 11 kg is low. Further, the calculations on which the criticality safety analysis is based assumed each can contains 12 kg of uranium metal, 33% more than the average can, and a full kg more than the maximum can. This is highly conservative, because the lower the uranium mass per can, the more cans that must be close together before a criticality is possible. With an average can containing 9 kg of uranium, rather than a person carrying two cans, holding them together, and placing them in contact with HEU-bearing equipment, he would instead have to carry three or more. At 9 kg (~20 pounds) per can, three cans weigh approximately 60 pounds, which obviously presents a considerable physical impediment to such an action occurring. It is also important to realize that for criticality to occur the equipment that the containers are brought close to must contain a significant amount of uranium. Much of the equipment in the BPF operates in batch mode, and therefore is sometimes empty, or in the process of being loaded, with only a small amount of uranium present. All of these factors show that a criticality accident would be unlikely even if two containers and HEU-containing equipment were brought into contact.

Recall above that the ISA analysis assumes that only three violations have to occur for a criticality accident to be possible. Considering the above discussion, we can add the following to that list:

1. The containers must be brought in close contact such that all are within ½ inch of each other. This action certainly has a lower probability than the simple action of violating the 12 inch spacing requirement.

2. <u>Each</u> of the containers must contain 11 kg of material <u>and</u> the HEU-containing equipment must contain a significant amount of material. As discussed above, this is very unlikely. It is also very unlikely that an operator would carry three containers at once.

There are other such conservatisms that could be considered, but the point has been made. The conclusion is that multiple low probability events that were not included in the ISA's assessment of the -4 Controlled Likelihood Index would also have to occur simultaneously before criticality would be possible as a result of this accident scenario. Thus, the likelihood of this scenario is at least an order of magnitude lower than even the "highly unlikely" determined by the ISA.

II. Backflow of Fissile Solution Into Plant Air System

Intervenors' brief cites two criticality accident sequences involving pump seal failure. Intervenors' Pres. at 30. These are accident sequences in the U Metal Dissolution Process Area:

4.1.28 Pump seal fails (ISA likelihood index of -5)
4.1.29 Pump seal fails (ISA likelihood index of -5)

The accident sequence Backflow-2 (ISA likelihood index of -5), in the OCB Uranium Recovery Process Area (cited in Intervenors' Pres. at 30), is also very similar to the two pump seal failure accident sequences discussed here.

The particular accident sequences referenced by the Interveners refer to HEU backflow into the Plant Air supply system. In such an accident, HEU would flow from a favorable geometry column, in which HEU solution is processed or stored, back through a Plant Air supply line into the Plant Air supply system. It should be noted that backflow into other utilities or into chemical supply systems would require the occurrence of very similar events and control failures. Thus, this discussion is also applicable to potential accidents involving HEU backflow into those systems as well.

Some of the operations in the BLEU facilities utilize favorable geometry columns to store or process HEU solution. A favorable geometry column has a limited diameter, such that criticality is not possible, regardless of the column height or the concentration of uranium in the solution. This is a very robust passive engineered control. In some cases these columns are serviced by utilities, such as plant air, or by chemical supply lines. The utility and chemical supply lines are very small in diameter and therefore of favorable geometry, but often lead to large tanks that are not of favorable geometry. Therefore, it is necessary to provide means to assure HEU solution will not backflow from the favorable geometry columns into the utility or chemical supply lines.

Figure 1 illustrates a simplified arrangement of a favorable geometry column that is supplied uranium-bearing solution from another favorable geometry source. The column contents are pumped out through the drain line using an air diaphragm pump that is supplied by the Plant Air system. An overflow line on the column is vented to atmosphere, thereby assuring that the column contents are normally at atmospheric pressure.

In order for uranium-bearing solution to backflow into the Plant Air system, the solution pressure must exceed the pressure in the plant air system, and any barriers to flow must be removed. The accident sequence can be described as follows.

1. The initiating event is an uncontrolled addition of HEU process solution into the favorable geometry column. The transfer operation is controlled using a manual valve upstream of the favorable geometry column. Therefore, in order for the

initiating event to occur, either the valve must experience a mechanical degradation, or an operator must mistakenly leave it open. Such an event is expected to occur only a few times during the life of the facility.

2. The first enabling event is a failure of the overflow line to vent the favorable geometry column, thereby allowing the column to become pressurized. The overflow line is a robust passive engineered control. The only way it could fail is if it became plugged, but there are no solids in the system that could cause this to occur. Despite the highly robust nature of the overflow as a means of preventing backflow, and the resulting low likelihood of this event, it is not credited in the ISA.

3. The second enabling event is a failure of the Plant Air supply system, such that the pressure in the Plant Air lines reduces to near atmospheric levels. Variations in Plant Air pressure are expected, but failure of the system to very low levels is not a common occurrence, expected to occur a few times during the life of the facility. However, this low frequency is also not credited in the ISA.

4. The third enabling event is failure of the diaphragm on the drain line transfer pump, which removes the barrier between the process solution and the plant air line. Such a failure is expected to occur with a low frequency, due to periodic maintenance on the pump, and the requirement that the diaphragm material of construction be compatible with the chemicals being pumped. The combination of the initiating event (step 1) and this enabling event is conservatively assigned a frequency index of -1.

5. The first IROFS is a pressure sensor interlocked to a pneumatic valve. The pressure sensor has a set point of 70 psi. If the Plant Air supply pressure drops below 70 psi, the valve automatically closes. It must fail for the accident to occur. This active engineered feature is conservatively assigned a failure index of -2.

6. The second IROFS is a second, independent pressure sensor/interlocked valve. It must also fail for the accident to occur. It is also conservatively assigned a failure index of -2.

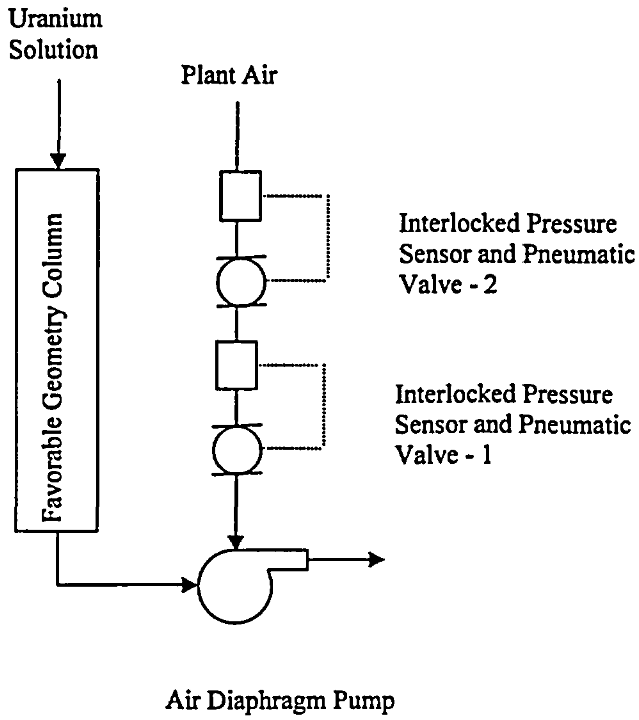

Figure 1

The ISA determined that this accident sequence had a Controlled Likelihood Index of –5, based on the fact that both IROFS, with Effectiveness of Protection Indices of –2, have to fail, and the initiating event/enabling event index is -1. The likelihood of occurrence of the other two enabling events is <u>not</u> credited in the assessment. There is absolutely no commonality between the initiating event and any of the three enabling events. Therefore, the probability that the initiating event and three enabling events would all occur concurrently is so low that the accident sequence probably is not even credible.

Furthermore, that is not the end of the conservatism of the likelihood assessment. The above discussion is focused on prevention of backflow. It is also important to understand that backflow of uranium solution into the Plant Air system does not assure a criticality will occur. The non-favorable geometry tanks in the Plant Air system are large cylindrical shapes. Backflowing solution will first spread out to form a thin slab in the bottom of the tank. Criticality would only be possible as the depth of that slab became significant (the exact amount depends on many factors, but at least 3 inches). The point is, a significant volume of uranium solution would have to backflow all the way into the non-favorable geometry tanks before criticality would be possible. This would take a significant amount of time, during which it is likely an operator would notice the problem and close a valve to terminate the backflow.

Therefore, when the likelihood of the accident sequences is assessed more completely and more realistically, one can see that the likelihood is so low as to be non-credible. The analysis in the ISA was terminated once the Controlled Likelihood Index for the accident was found to not exceed –4 because that is all that is required under 10 C.F.R. Part 70. A license applicant only has to show that potential criticality accidents (whose consequences are presumed by regulation to be "high") are "highly unlikely." There is no requirement that the ISA go further to show that they are not credible or to show how far below the point of credibility their likelihoods really are.

III. Spill of Uranium Solution in the U/Al Dissolution Area

Intervenors' brief cites a criticality accident sequence involving a spill of HEU solution: Spill-2A in the U/Al Dissolution Area (ISA likelihood index of -5). Intervenors' Pres. at 30.

The accident sequences Spill-6 (ISA likelihood index of -5), in the OCB Precipitation Process Area, and Spill-6 (ISA likelihood index of -4) in the OCB Dryer/Calciner Process Area (cited in Intervenors' Pres. at 30), are not specifically discussed below but they are very similar to the HEU spill discussed here:

The accident sequence referenced by the intervenors involves a spill of HEU solution in the BPF. Uranium solution in the U/Al dissolution area is created in dissolvers, stored in columns and transferred between columns and dissolvers in stainless steel tubing. Dissolvers, columns, and tubing are all of favorable geometry – i.e., because of their size and shape, criticality is not possible regardless of uranium concentration. Uranium solution could leak from a ruptured vessel, a damaged pump, or a failed valve. Such an initiating event might occur several times during the plant lifetime, and therefore is assigned a -1 Initiating Event Failure Frequency Index.

The accumulation of uranium solution on the floor, however, cannot cause a criticality, because the solution spreads out into a very thin slab. The only criticality concern associated with uranium solution leaks arises if an operator were to attempt to use a non-favorable geometry container to catch the leaking solution. It should be noted that all operators are repeatedly trained on and exhibit a very high awareness of the implications of using non-favorable geometry containers with HEU.

Two controls are in place to prevent the use of a non-favorable geometry container for collecting leaking uranium solution in the U/Al dissolution area of the BPF. First, non-favorable geometry containers are not permitted to be brought into the BPF unless they have lids securely attached in a manner that would prevent liquids from entering the container. The very few exceptions must be specifically approved in writing by the criticality safety department, and are generally specific-use items that are in the facility for only a brief period of time, and which are not left open and unattended. An example is the use of a large plastic bag to collect water from a test of the safety showers. Since non-favorable geometry containers are not available in the BPF, they are not available for use in collecting leaking uranium solution in the U/Al dissolution area.

The second control is a requirement to use only two-liter or smaller bottles to handle fissile solution in the BPF. It takes at least 4 liters of HEU solution to support criticality, so use of two-liter or smaller bottles assures criticality will not occur. Therefore, two administrative controls must be violated before a non-favorable geometry container could be used to collect leaking uranium solution in the U/Al dissolution area. Each of these administrative controls is assigned an Effectiveness of Protection Index of -2. When combined with the initiating event index of -1, the total Controlled Likelihood Index for the sequence is -5. This is less than the -4 index necessary to comply with 10 C.F.R. § 70.61, and therefore demonstrates defense in depth for this accident sequence.

There are other salient factors that are not accounted for in the ISA analysis that make the likelihood of the accident sequence even lower than what the ISA suggests. All fissile material operators receive basic training in criticality safety. One of the basic tenets that is taught is that large containers and HEU solution lead to criticality. NFS has several HEU facilities at the Erwin site and a very long history of processing HEU solutions, so operators are keenly aware of the non-favorable geometry container issue. This makes them much less likely to violate the limits and (1) bring such a container into the BPF, and (2) use it to contain a uranium solution spill.

Furthermore, there is a scarcity of non-favorable geometry containers at the NFS site. The most common non-favorable geometry container is a 55-gallon drum. These are used for contaminated trash, and operators are well aware that when in the BPF they must have their lids securely attached.

Finally, it is unlikely a single operator would be alone in trying to contain an HEU solution leak. It is most probable that he would either ask for help, or that someone else would notice the leak and come to assist. It is even more highly unlikely that several operators would fail to recognize the danger in utilizing a non-favorable geometry container for leak collection in the U/Al dissolution area.

IV. Excess Uranium Ingots in Enclosure

Intervenors' brief (Intervenors' Pres. at 30) cites two accident sequences involving the placement of too many HEU/Al ingots in enclosures in the U/Al Dissolution Area:

Enclosure-2a (ISA likelihood index of -4)
Enclosure-2b (ISA likelihood index of -4)

The first step in the U/Al dissolution process in the BPF is to load ingots made of a Uranium/Aluminum alloy into the dissolvers. The dissolvers are stainless steel pipes in a vertical orientation. The top of each dissolver penetrates the bottom of a loading enclosure. The enclosure is constructed of sheet metal with a clear plastic front for viewing. Gloveports on the glass front allow operators to perform manual manipulations inside the enclosure.

Criticality safety for an enclosure relies on limiting the mass of uranium present in the enclosure at any one time. Calculations indicate that more than six containers of U/Al ingots (i.e., six ingots or over 42 kg U/Al) would have to be present inside the enclosure before criticality would be possible. An active engineered control is utilized to assure not more than four containers will be in the enclosure at any one time. Further, administrative controls require no more than one ingot be in the enclosure at any one time. These two controls are described in detail below.

The only way to bring a container into the dissolution enclosure is through a small airlock that is sized to only accept one container at a time. The airlock has a scale that is connected to a Programmable Logic Controller (PLC). The dissolver and its enclosure are mounted on a load cell, which is also connected to the PLC. The system works as follows:

1. The door to the air lock from outside, and the door from the airlock to the enclosure, are each fitted with magnetic locks. The PLC allows only one of these doors to be open at a time.

2. The PLC is programmed to allow a maximum mass of 28 kg in the enclosure/dissolver. This corresponds to four U/Al ingots at 7 kg/ingot.

Loading of U/Al ingots into the enclosure/dissolver is as follows:

1. The air lock door is opened and a U/Al container is brought into the air lock and placed on the scale. Note that the enclosure door remains locked.

2. The PLC senses the mass of the container on the scale, and the mass of the containers already in the enclosure/dissolver, and adds the two together. If the total mass does not exceed 28 kg, the PLC unlocks the enclosure door.

3. The operator opens the enclosure door and brings the container into the enclosure. Note that the air lock door is locked by the PLC while the enclosure door is open.

4. By administrative procedure the operator is required to remove the ingot from the container and place it into the dissolver before he brings another container into the enclosure.

These controls form highly robust barriers to prevent the mass limit of 28 kg from being exceeded. Both the active engineered control and the administrative control are assigned Effectiveness of Protection Indices of -2, although a -3 is probably justifiable for the active engineered control. This results in the -4 Controlled Likelihood Index assignment in the ISA. Such an index indicates a highly unlikely accident. Note, however, that this assessment is conservative because there is no credit taken for the unlikelihood of an operator attempting to place more than four U/Al ingots into an enclosure in the first place.

The controls discussed above prevent more than four U/Al ingots from being present inside an enclosure at one time. As previously discussed, the Nuclear Criticality Safety Evaluation (NCSE) demonstrates that at least six ingots would have to be inside the enclosure before criticality is possible. However, the calculation that led to that conclusion was extremely conservative, as can be seen from the table below, and thus it is even more unlikely that the mass of U/Al in an enclosure would even approach criticality.

	Actual	Modeled
^{235}U Enrichment	65%	100%
U/Al Density	3.96 g/cc	10 g/cc
U/Al Mass per Ingot	Max 6.8 kg, Avg 5.0 kg	16.9 kg
Mass Percent U in Ingot	18.5%	30%

The amount of conservatism in the calculation is enormous. If actual parameters were used in the model, it is likely that the results would indicate that more than 12 ingots would have to be present for criticality to be possible.

Another conservative feature of the criticality model is in the assumption that the enclosure is completely surrounded on all sides by a thick layer of water. The purpose of this assumption is to assure that any neutron reflection that might occur from the bodies of operators standing around the enclosure is conservatively accounted for. However, actual neutron reflection from human bodies standing around the enclosure would not come close to that provided by the presence of the thick, continuous layer of water assumed in the model. A realistic reflection model combined with modeling of realistic ingot parameters would probably demonstrate that more than 18 ingots would be required before criticality would be possible.

The final conservatism that must be considered is the spacing between ingots. The calculations assume the ingots are arranged in a tight-fitting array, with each ingot in contact with the other. It is more likely ingots would be randomly distributed inside the enclosure. This would further increase the number of ingots necessary for criticality to be credible.

The ISA reported a -4 Controlled Likelihood Index for this accident sequence, based on the two controls discussed above. However, as has been shown, the actual safety margin is <u>much</u> larger, since so many ingots would have to be brought into the enclosure that criticality is probably not even credible.

Again, when the likelihood of the accident sequences is assessed more completely and more realistically, one can see that the likelihood is so low as to be non-credible. The analysis in the ISA was terminated once the Controlled Likelihood Index for the accident was found to be –4, i.e., highly unlikely, because that is all that the regulations require. There is no requirement that the ISA go further to show that the sequences are not credible or to show how far below the point of credibility they really are.

V. Increase in Uranium Concentration Due to Low-Temperature Induced Crystalization in the Transport Tank

Intervenors' brief (Intervenors' Pres. at 31) cites two criticality accident sequences in the UN Receipt Area of the Uranyl Nitrate Building (UNB) caused by the cold weather-induced concentration of HEU solution in the transport tank:

1.7.1	High U in TK-10 (ISA likelihood index of -4)
1.18.1	High U in TK-10 Feed Line (ISA likelihood index of -4)

The UNB houses a number of large tanks designed to safely store low-enriched uranyl nitrate solution (LEUN). The LEUN is transferred to the UNB from two different sources: (1) the NFS downblending facility, i.e., the BPF, and (2) the Savannah River Site (SRS) in Aiken, SC. The LEUN is brought from SRS in tanks mounted on a flatbed truck trailer. The solution shipped from SRS is the subject of these accident sequences.

The LEUN is loaded into a LR-230 Shipping Container at Savannah River Site. In order to meet the license requirements for the LR-230, SRS limits the uranium concentration in the solution to 125 g/L. The LR-230 is then brought by truck to the UNB at NFS' Erwin facility. At the UNB the LEUN solution is transferred from the LR-230 into the receipt tank (TK-10). The uranium concentration operating limit for TK-10 (and indeed for all of the tanks in the UNB) is 210 g/L, and calculations show that criticality is not possible unless the uranium concentration exceeds 280 g/L. There is a very large margin between the 125 g/L limit for the LR-230 and the 210 g/L limit for the UNB.

One area of concern is an increase in uranium concentration that occurs during transport of the LR-230 as a result of low temperature-induced crystallization. Experiments performed at NFS indicate that if the LEUN solution is exposed to below-freezing temperatures for a prolonged period of time, the uranium becomes concentrated in a liquid phase at the bottom of the container, with frozen water floating above. Experiments were performed with LEUN solution with uranium concentrations of 100 g/L and 190 g/L. After being subjected to -18°C (-0.4° F) temperature for 6 days, the uranium concentration in the liquid phase at the bottom had increased to 334 – 364 g/L. Since this concentration is greater than the minimum 280 g/L required for criticality to be possible, controls were established to assure such liquid would not be introduced into TK-10.

The LEUN is transferred to the LR-230 to TK-10 by pressurizing the LR-230 with compressed air. This causes the liquid to flow out of the LR-230, through a pipe, and into TK-10. The pipe is fitted with two independent temperature interlock systems that will automatically terminate the transfer if the solution temperature is ≤ 35°F (1.7°C). The maximum temperature at which uranyl nitrate solution will freeze is 32°F. Therefore, the interlocks assure that the solution has not been concentrated by crystallization.

The first temperature interlock consists of a thermocouple that, when activated by the low temperature condition, causes a valve on the transfer line to close, and also causes the compressed air line to be shut off and vented. Either one of these actions alone is sufficient to terminate the transfer. This highly robust engineered control is designated as an IROFS. A -2 Effectiveness of Protection Index was assigned to this IROFS in the ISA, although a -3 could easily be justified.

The second temperature interlock consists of a thermocouple that, when activated by the low temperature condition, causes a valve on the transfer line to close. This second temperature interlock utilizes a different thermocouple and a different valve from the first interlock, and is therefore completely independent from the first. This engineered control is designated as an IROFS and assigned an Effectiveness of Protection Index of -2 in the ISA. Again, a -3 index could easily be justified.

The Controlled Likelihood Index of -4 listed in the ISA is obtained by summing the failure indices for the two IROFS. Note that a -6 index could easily be justified by assigning failure indices of -3 to each of the robust active engineered controls.

Furthermore, the discussion to this point has focused on controls that prevent solution that has been frozen during shipment from being pumped into TK-10. It is important to consider the likelihood of such solution ever being present.

Procedures at SRS require that the LEUN solution be maintained at 59°F or higher when loading into the LR-230. The average daily temperature during the coldest month of the year (January) in Erwin, TN, is 35.7°F, or 2.1°C (note that Aiken, SC experiences much milder winter conditions than does Erwin). Therefore, freezing of the LEUN in the LR-230 most likely would not occur even if the LR-230 was left outside in a loaded condition in Erwin for an extended period of time. Of course, in reality the LR-230 is normally unloaded immediately upon arrival at the UNB. Also, the LR-230 is heavily insulated, and the large liquid volume has a very high heat capacity. Therefore, it is highly improbable that the initiating event for the accident sequence, freezing of the LEUN in the LR-230, would ever occur.

Another factor to consider for this accident sequence is the fill condition of TK-10 prior to initiating the transfer. Assume that neither of the temperature interlocks function as designed. If there is normal-condition solution (uranium concentration of 125 g/L or less) already in TK-10 when the transfer is initiated, then the added (higher concentration) solution will mix with the existing solution, resulting in a lower effective concentration. The maximum concentration of the crystallized LEUN is approximately 364 g/L, and the concentration below which criticality is not possible is 280 g/L. Therefore, if the volume of normal condition solution in TK-10 prior to initiating the transfer is at least 55% of the volume in the LR-230, then the mixed solution after the transfer is completed will have a uranium concentration of 280 g/L or less, and criticality will not be possible.

The temperature conditions inside TK-10 need also be considered. All of the UNB tanks, including TK-10, are inside a climate-controlled building, with temperatures maintained at a minimum of 65°F. Again, if both temperature interlocks failed to function as designed, and cold, crystallized LEUN is added to the tank, it will mix with the warm liquid already inside the tank. This will raise the temperature of the mixture above the freezing point, causing the crystallized UN to dissolve back into solution, which would reduce the concentration below its crystallized level.

This accident sequence is assigned a -4 Controlled Likelihood Index in the ISA summary, but clearly the real likelihood of the event occurring is extremely low, probably low enough to make it not credible. The -4 likelihood index: (1) is based on a -2 Effectiveness of Protection Index assignments for each of the two highly robust active engineered controls, where -3 indices could be easily justified; (2) does not account for the likelihood of the required initiating event, which is extremely improbable; and (3) does not account for mitigating factors, such as dilution and warming/redissolution of crystallized LEUN if it were to be added to TK-10. The analysis in the ISA was terminated once the accident was found to be highly unlikely, but in fact these accident sequences are much more unlikely than even what the ISA indicates.

VI. High Uranium Concentration Due to Precipitation in TK-10

Intervenors' brief (Intervenors' Pres. at 31) cites another criticality accident in the UN Receipt Area of the UNB concerning high uranium concentration due to precipitation in the receipt tank (TK-10).

1.12.2 High U in TK-10 (ISA likelihood index of -5)
1.26.3 TK-10 High U due to precipitation (ISA likelihood index of -4)

As discussed above, TK-10 is a large receipt tank in the UNB that contains LEUN solution. Criticality safety in this tank is achieved by controlling both the enrichment and the concentration of the uranium in the solution. As was previously discussed, criticality in this tank is not possible if the uranium concentration is kept below 280 g/L. The solution received in this tank under normal conditions has a uranium concentration of 125 g/L or less.

While it was not definitively known to be a potential cause of an accident, chemical precipitation was investigated to determine if it could potentially increase the concentration of uranium solution already in the TK-10 tank. There are a limited number of chemicals that will cause precipitation of uranium from uranyl nitrate solution. None of these chemicals are piped to or otherwise utilized in the UNB, although several of them are used at other locations on the NFS site. Sodium hydroxide, calcium hydroxide, and ammonium hydroxide are by far the most commonly utilized of these onsite precipitation agents. The only other precipitating agent utilized at NFS is H_2O_2, but its use is extremely limited. Despite the fact that none of these chemicals are piped to the UNB, and they are expressly forbidden from being brought into the facility, the consequence of precipitation by these agents was investigated. Uranyl nitrate solutions with uranium concentrations in the range of 150 – 350 g/L were precipitated with ammonium hydroxide, sodium hydroxide, and H_2O_2 in a series of experiments. Precipitation was observed in all cases, but this did not cause the uranium concentration in the solutions to change. This result indicates that uranium precipitation in these solutions does not represent a means for increasing uranium concentration, and therefore is not a criticality concern.

Despite this result, explicit controls were implemented to prevent introduction of precipitating agents into TK-10. There are two different accident sequences that could lead to precipitating agents entering TK-10: (1) Precipitating agents are transferred from a container in the receipt station into TK-10, or (2) Precipitating agents are pumped from the spill basin sump or the sink into TK-10. The initiating event for the first of these accident sequences is for a vessel other than the LR-230, containing a precipitating agent, to be brought into the Truck Bay for connection to TK-10. Since the Truck Bay is exclusively utilized for uploading LEUN from the LR-230 to TK-10, this initiating event is conservatively assigned an initiating event frequency index of -1, which corresponds to an event expected to occur during the facility lifetime. Were this initiating event to occur, the operators would not be able to connect the vessel to the transfer system due to the first IROFS, which is a passive engineered feature that consists of unique fittings on the feed line that allow connection only to the LR-230 vessel. This highly robust passive engineered feature is assigned a failure frequency index of -3, although a -4 index could probably be justified. The second IROFS is an administrative control that allows operators to only mate the TK-10 transfer line to an LR-230 vessel. Operators are trained on this requirement and observe it on a daily basis. A -1 failure frequency index is assigned to this administrative control The total likelihood index for the sequence is -5.

The second accident sequence results in addition of precipitating agents via the spill basin sump or the sink into TK-10. The first IROFS to protect against this sequence is a pH sensor interlocked to the pump. Precipitating agents are basic and thus have a pH greater than 9. Therefore, the system is designed to cause the pump to be turned off, thus terminating the transfer, if the pH of the solution exceeds 9. This is a robust active engineered control that is assigned a failure frequency index of -2, although a -3 could be justified. The second IROFS is an administrative control that prohibits the use of precipitating agents in the UNB. There is a large margin of safety associated with this administrative control, because calculations show a minimum of 6.4 gallons of a highly effective precipitating agent (50% NaOH) is required to precipitate a sufficient mass of uranium for criticality to be feasible. Any cleaning agents that might be used in the UNB and which contain an unrecognized precipitating agent would (1) be much less effective as a precipitating chemical, and (2) be used in quantities much less than 6 gallons. Therefore, the administrative control is robust and a -2 failure frequency index is assigned. The total likelihood index for this accident sequence is -4.

It is important to understand that the accident sequence is the presumed increase in uranium concentration following the introduction of a precipitating agent in TK-10. As was discussed above, introduction of a precipitating agent was shown via experiment <u>not</u> to increase the uranium concentration in these solutions. Thus, the experimental data suggests that concentration increase via precipitation is not possible and therefore is not a criticality concern. The analyst chose to prevent addition of precipitating agents to provide defense in depth. Nevertheless, based on the experimental data, the likelihood of a criticality accident occurring via this accident sequence is extremely low and it is probably not credible.

VII. Water Introduced Into a Moderation Controlled Area to Fight a Fire

Intervenors' brief (Intervenors' Pres. at 30) cites another criticality accident in the Oxide Conversion Building (OCB) concerning the use of water to fight a fire in the ModCon Area:

External water into blending system – water used to fight a large fire inside the ModCon area (ISA likelihood index of -5)

One area in the Oxide Conversion Building (OCB) contains large, non-favorable geometry tanks (vessels V-35 and V-36) that process dry UO_2 powder. Introduction of liquid water into one of these tanks could, if a sufficient mass of UO_2 powder was present, result in a criticality accident. For this reason, the area in which these tanks are contained is defined as a Moderation Controlled Area, or ModCon Area. Robust design features are utilized to prevent water or other liquid moderating materials from entering the ModCon area. There are no liquid-bearing lines inside the ModCon area, and the area is covered by a double roof. Liquid detection instrumentation is present between the two roofs, to warn of a leak in the first roof. The area is enclosed to separate it from the non-ModCon areas and to prevent liquid from those areas from being sprayed into the ModCon area. Doors are self-closing, and are not permitted to be propped open. In addition to these engineered design features, administrative controls prevent operators from bringing water into the ModCon area, and also prohibit the use of water for fighting fires inside the ModCon area.

The initiating event for this accident sequence is ignition of a fire inside the ModCon area. A frequency index of -1. was assigned to this initiating event in the ISA, corresponding to an event expected to occur during the life of the facility. Such a frequency would be consistent with anticipated fire frequency in a normal industrial facility. However, the ModCon area is specially designed and maintained to assure a fire does not occur. Therefore, the -1 initiating event frequency index is very conservative.

The oxide blending system is a closed system, with the powder contained within sealed vessels. The use of water to fight a fire in this area is only a concern if the water can enter one of the vessels. The only way this could happen would be if the fire caused degradation of the stainless steel components of the oxide blending system. The Fire Hazard Analysis (FHA) for the OCB determined that the bounding fire in the ModCon area would result in a maximum temperature of 1850°F, insufficient to significantly degrade the stainless steel components comprising the oxide blending system. Therefore, a fire that is larger and more severe than the bounding fire analyzed in the FHA would have to occur in the ModCon area before a criticality could result from this accident sequence. A more severe fire could conceivably occur if there is more combustible material in the area than was analyzed in the FHA. Therefore, an IROFS was established to prevent the storage of significant amounts of combustible material in the ModCon area. This administrative control is routinely observed and stressed to operators during training. Therefore, a -2 failure frequency index is assigned to this IROFS.

The use of water to fight fires in the ModCon area is prohibited by an administrative control that is also designated as an IROFS. This control is easy for operators to remember since they are so conditioned to the concept of keeping water and other liquid moderators out of the ModCon area. There are also very clear postings at the entrances to the ModCon area reminding personnel to not use water to fight a fire. Portable dry chemical fire extinguishers are available in the area for use in putting out a fire. For these reasons a -2 failure frequency index is assigned to this IROFS. The total likelihood index for the accident sequence is -5.

The occurrence of a fire in the ModCon area with failure of both of the IROFS would not necessarily result in a criticality accident. First, consider that the oxide blending system operates in batch mode, so the vessel is not always full. The fire would have to occur when the vessel contained a significant amount of UO_2 powder. More importantly, fire-induced degradation of the stainless steel components that make up the oxide blending system does not automatically assure water will be introduced into the non-favorable geometry vessels. The degradations could be at a location that makes entry of water difficult. The degradations could be at locations away from the base of the fire, such that fire fighting water would not be aimed toward them. Or the fire fighting personnel could purposefully avoid spraying water toward the non-favorable geometry vessels, because of their training and understanding of the potential consequences. When these issues are considered the likelihood of a criticality accident occurring as a result of this accident sequence is significantly lower than that indicated from the ISA analysis.

VIII. Wet Powder Discharged From Calciner into Non-Favorable Geometry Vessels in ModCon Area

Intervenors' brief (Intervenors' Pres. at 30) cites another criticality accident in the Oxide Conversion Building (OCB) concerning the discharge of wet UO_2 powder from the calciner and into non-favorable geometry vessels in the oxide blending area.

Ammonium diuranate (ADU) is precipitated from uranyl nitrate solution. The wet solids are then passed through a centrifuge and then an electrically heated drier. Both of these steps remove water from the ADU. The dried ADU enters the calciner where it is heated at a high temperature in a reducing atmosphere. This removes any remaining water from the powder and also induces a chemical reduction reaction, which changes the uranium-bearing species from ADU to UO_2. The dry UO_2 is discharged from the calciner into favorable geometry receiving vessel V-33 or V-34, and subsequently is transferred to non-favorable geometry vessels V-35 and V-36. Criticality safety in the non-favorable geometry vessels relies on the powder being dry.

The initiating event for the accident sequence is a failure in the upstream process equipment that results in the discharge of wet UO_2 powder from the calciner. This could be a failure in the centrifuge, the electric drier, the calciner, or some combination of these. It should be noted that even if some type of failure resulted in the introduction of ADU solids with a high moisture content into the calciner, it is unlikely that wet UO_2 solids would be discharged, because the calciner operates at a very high temperature (1,450°F), and the solids have a long residence time in the calciner (100-120 minutes). This long residence time in the calciner also would give operators plenty of time to notice something amiss with the upstream equipment. For these reasons the assignment of a -1 frequency index to the initiating event is very conservative. A lower assignment could probably be justified.

Two independent engineered features are used to assure that if wet powder is discharged from the calciner it does not end up in the non-favorable geometry vessels V-35 and V-36. The first of these IROFS functions as follows. UO_2 powder falls from the end of the calciner into favorable geometry vessel V-33 or V-34, passing through a counter-current stream of dry nitrogen gas. Any residual moisture in the discharged powder will diffuse into the dry nitrogen gas, thereby raising its dew point. After passing through the UO2 powder, the nitrogen gas enters a moisture analyzer. The moisture analyzer is interlocked to valves on the transfer lines from vessels V-33 and V-34 to vessel V-35, and to the vacuum transfer blower. If the moisture content of the nitrogen gas exceeds the setpoint, closure of the valves prevents transfer of the powder to the non-favorable geometry vessel. As a second precaution, the vacuum transfer blower is disabled, which makes it impossible to transfer the powder to vessel V-35 even if the valves remain open. This robust active engineered control is assigned a failure frequency index of -2, but a -3 index is clearly justified.

The second IROFS is a second independent moisture analyzer that operates in a similar manner to that described above, but that measures moisture in the UO_2 power in vessel V-33, as opposed to the powder falling from the calciner into vessel V-33. This IROFS is completely independent from the first. It is also assigned a failure frequency index of -2, although a -3 is clearly justified.

A criticality could result from this accident sequence from only the failures listed in the ISA. However, the failure frequency indexes assigned in the ISA to the initiating event, and to the two IROFS, was very conservative. The likelihood of the initiating event occurring is low, and the long residence time in the calciner makes it likely that operators would have time to notice the failure before wet powder could exit the calciner. The two active engineered controls are robust. Therefore, a -8 likelihood index could be justified for this sequence, as opposed to the -5 listed in the ISA.

IX. Water Enters Blending System Via Compressed Air System

Intervenors' brief (Intervenors' Pres. at 30) cites a criticality accident in the Oxide Conversion Building (OCB) concerning water intrusion into dry UO_2 powder in the oxide blending system via the compressed air system:

Water enters blending system through compressed air system – internally supplied air (ISA likelihood index -5)

UO_2 falls from the calciner into either vessel V-33 or V-34. Vessel V-34 is used to further oxidize UO_2 to U_3O_8. This is a batch process that utilizes 20-50 kg of UO_2 powder per batch. The powder is first heated to 250°C using an external 4 KW heater. Next, compressed air heated to 160°C is introduced into the vessel. This initiates the oxidation reaction, which is exothermic and self-sustaining. The external heater is turned off, and the heat of reaction raises the temperature of the powder to a maximum 500C. When the reaction is complete the powder is cooled with nitrogen gas before being discharged in the transfer station.

The accident sequence is concerned with the presence of water in the compressed air. If a significant quantity of water were present in the air, it could be absorbed into the powder. Vessel V-34 is of favorable geometry, so wet powder is not a criticality concern in that vessel. However, if that wet powder were transferred to non-favorable geometry vessel V-35 or V-36, a criticality could occur.

The compressed air system consists of a compressor that pressurizes air inside a 120 gallon reservoir tank. The outlet line to the compressed air system exits the reservoir from the vertical center of the tank. Therefore, at least 60 gallons of water would have to accumulate in the tank before water could flow down the compressed air line. A purge valve on the bottom of the tank automatically opens once every minute. Any water that accumulates in the tank via condensation or any other mechanism is drained via this purge valve.

A robust air drying system is employed on the compressed air line as it exits the reservoir. This is a multistage system, with the first stage consisting of a moisture separator with an electronic drain valve. Downstream of the separator the air flows through a coalescing prefilter, which removes liquid water, oil, and particles 0.01 microns and larger with a 99.999% efficiency. The air then enters the bottom of an active dessicant vessel. Moisture is absorbed in the dessicant as the air flows up through the vessel. Finally, the air passes through a one micron after filter.

The initiating event for the accident sequence is that water condenses inside the compressor reservoir tank and enters the compressed air line. This initiating event is assigned a frequency index of -1, corresponding to an event expected to occur during the facility lifetime. This is a very conservative assignment. Consider that more than 60 gallons of condensate would have to accumulate in the reservoir, and that the active-engineered control (automatically-actuated purge valve) would have to fail. A lower frequency index is probably justified.

The first IROFS is the drying system, and it is conservatively assigned a failure frequency index of -2 in the ISA. This is a robust passive engineered system for which a -3 index could easily have been justified, and a -4 is probably justifiable. The second IROFS is the moisture analyzer as previously described in Section IX above. This is a robust active-engineered control that was assigned a -2 index in the ISA, but for which a –3 index is probably justifiable. The total likelihood index for the accident sequence is -5 as assigned in the ISA, but clearly an index of -7 or lower could have been justified.

The ISA did not credit the fact that the exothermic oxidation reaction that occurs in vessel V-34 causes the UO_2 powder to achieve a maximum temperature of 500°C. Any water that flows into the vessel from the compressed air system would be vaporized by the high temperature of the powder, and then swept away in either the air stream or the subsequent nitrogen purge stream. Therefore, even if both IROFS failed, it is not likely that the powder exiting vessel V-34 would be wet.

In summary, the risk of wet powder entering non-favorable vessels due to water in the compressed air line fed to vessel V-34 is much lower than indicated by the ISA due to the (1) conservative assignment of failure frequencies to the IROFS in the ISA, and (2) the fact that some preventative features are not credited in the ISA.

X. Too Much Dry Powder in Blender

Intervenors' brief (Intervenors' Pres. at 30) cites a criticality accident in the Oxide Conversion Building (OCB) the presence of too much dry powder added to the blender resulting in too large of a mass of water in the blender:

Too much dry powder in blender results in > 15.8 kg water (ISA likelihood index -5)

Blender vessel V-36 is a large, non-favorable geometry vessel. Criticality in this vessel is prevented by keeping the moisture content in the UO_2 very low. Under normal operating conditions the moisture content of the powder is approximately 3,000 parts per million (ppm). The safety limit is 5,000 ppm and is ensured by the moisture analyzers and other controls discussed for the scenarios in Sections VIII and IX. In reality, criticality cannot occur in the blender unless the moisture content of the powder exceeds 27,500 ppm, provided the moisture is uniformly distributed. Redistribution of moisture could theoretically occur as a result of evaporation, condensation, and re-absorption. Evaporation could occur as the powder is heated via mechanical agitation (blending) or by a chemical oxidation reaction (burnback). Evaporated water could rise into the headspace and then condense on the inner surface of the stainless steel vessel, and drip back into the powder. This theoretical accident scenario is analyzed in a highly conservative manner: it is assumed that when the water drips back into the powder, it is somehow concentrated and mixed with just the right amount of powder that forms a spherical region inside the blender. Under these conditions, it is theoretically possible for a criticality to occur if more than 15.8 kg of water is present. At the upper limit of 5,000 ppm water, 3,160 kg of UO_2 powder would contain 15.8 kg of water. To provide margin the blender is limited to a maximum UO_2 powder mass of 2,500 kg. This accident sequence is concerned with addition of more than 2,500 kg of UO2 powder to the blender.

The accident sequence requires non-uniform distribution of the moisture in the powder. In order to prevent this, procedure requires that when there is more than 1,000 kg of powder in the blender it must be operating (blending) or the nitrogen purge must be flowing. Typically the blender is placed into operation and nitrogen flow is initiated as soon as UO_2 powder is added to it. The nitrogen purge sweeps any evaporated water vapor from the head space and out of the blender, thus preventing it from dripping back into the powder. Blending assures uniformity of the powder, including any condensed water. Either of these actions is sufficient to prevent the non-uniform redistribution of moisture in the UO_2 powder, and thus prevent criticality. These administrative requirements are performed routinely and are part of the normal operation of the system (as opposed to being something special required only for safety). Also, nitrogen purge flow is monitored and an alarm is indicated if that flow is lost. Therefore, a -2 failure frequency index was assigned in the ISA for the failure of this IROFS (blending or nitrogen purge).

The initiating event is the addition of more than 2,500 kg of powder to the blender vessel V-36. Powder is transferred from receiving vessel V-33 or oxidation vessel V-34 to weigh hopper V-35. The weigh hopper is mounted on load cells. When the weigh hopper contains 800-1000 kg of UO_2 powder, it is discharged to the blender vessel V-36. A blend batch typically consists of three transfers from the weigh hopper. UO2 powder is discharged from the calciner at a rate of 50 kg/hr. Therefore, it takes approximately 50 hours to accumulate a blend batch, and the weigh hopper is dumped every 16 – 20 hours. This slow, deliberate accumulation makes it easy for operators to keep track of the mass in the blender and comply with the 2500 kg limit. The initiating event, loading of more than 3, 160 kg of UO_2 powder into the blender, is a failure of the IROFS that prohibits more than 2,500 kg of UO_2 powder in a blend batch. Because the loading process occurs in such a slow, deliberate manner, and because there is a significant margin between the limit (2,500 kg) and the minimum mass required for the even to proceed (3,160 kg), the assignment of a -2 failure frequency index in the ISA is conservative.

Recall that the 3,160 kg minimum UO_2 powder mass required for the accident sequence to occur is based on the UO_2 powder having the maximum allowed moisture content of 5,000 ppm. Normal operating conditions are for the powder to have a moisture content of approximately 3,000 ppm. With this normal moisture content, at least 5,000 kg of powder would have to be loaded into the blender for the accident sequence to proceed. Therefore, if the powder moisture content is normal, the blender would have to contain twice the permitted mass of UO_2 powder, which would require dumping the weigh hopper three times, in order for the sequence to proceed. This is a further element of conservatism in the analysis.

Finally, consider the enabling events for this accident sequence. Heating of the powder, either via mechanical agitation or oxidation reaction has to provide sufficient heat to evaporate all of the water contained in the powder. All of that evaporated water has to find its way to the head space, where it condenses on the inner stainless steel surface of the blender, and drips into the powder. Then that condensed water must mix with a small fraction of the total powder volume, and that fractional volume must form a spherical shape inside the bulk powder volume. This last part of the enabling events is the least likely and may not be credible. The ISA assigned a -1 frequency index to the enabling event, corresponding to an event expected to occur during the facility life. This is a highly conservative assignment, given that the sequence of events may not be credible.

To summarize, the credibility of this accident sequence is dubious at best. Criticality is highly unlikely even if both IROFS fail. Several low probability events have to occur in addition to those credited in the ISA for the accident sequence to proceed.

XI. LEU Powder Added to Natural Dissolver

Intervenors' brief (Intervenors' Pres. at 31) cites a criticality accident in the Oxide Conversion Building (OCB) when LEU powder is added to the natural uranium dissolver:

LEU powder added to natural dissolver (ISA likelihood index -5)

This accident sequence was assigned a -5 index in Rev. 0 of the NCSE, which was reported in the ISA reviewed by Intervenors. However, the sequence was subsequently declared not credible in Rev. 1 of the NCSE, due to changes in the facility design. The discussion below is based on Rev. 1 of the NCSE and shows that the not credible designation is warranted.

Natural uranium in the form of uranyl nitrate solution is a secondary product produced in a separate part of the OCB for shipment to SRS. Natural uranium for preparation of the uranyl nitrate solution is shipped to the OCB in the form of UO_3 powder contained in 55 gallon drums. These drums have a diameter of 22½ inches and a height of approximately 26 inches. The natural uranium UO_3 is dissolved in a 500 gallon dissolution tank that is not of favorable geometry for LEU. Therefore, it is important to assure that LEU (which is the other product produced in the OCB) is not added to the natural uranium dissolution tank.

The natural uranium dissolution and processing area is located in a separate part of the OCB, away from LEU processing equipment. Natural uranium is charged to the dissolver by placing the 55 gallon drum inside the feed hood via a transfer crane. An enrichment monitor on the feed crane arm easily distinguishes between natural uranium and LEU, and disables the crane if LEU is detected, thus preventing it from even being loaded into the feed hood. The feed hood has a vertical clearance of 39 inches, almost two inches less than the height of the LEU product pail. This prevents the LEU product pail from standing upright in the feed hood.

The feed hood is also equipped with an enrichment monitor that can easily distinguish between LEU and natural uranium. If LEU is detected, the vacuum transfer system that is used to transfer powder from the container in the feed hood to the dissolver is automatically disabled.

Another engineered feature in the feed hood is the use of a proximity switch. The proximity switch is also interlocked to the vacuum transfer system. Containers smaller than 55 gallon drums (i.e., the LEU product pails) will not activate the proximity switch.

All containers in the OCB must contain a color-coded label. The color is different for natural uranium and LEU, providing operators with another cue to the identity of the material. Operators receive criticality safety training and understand the significance of adding LEU to a non-favorable geometry vessel.

No one in the industry utilizes a 55 gallon drum as a shipping container for LEU. Therefore, LEU from offsite cannot be shipped to the OCB in a 55 gallon drum. LEU also cannot be loaded into a 55 gallon drum inside the OCB because those drums do not fit inside the LEU loading enclosures in the OCB, and they are not permitted to be used for things such as spill cleanup inside the OCB.

Due to the multitude of controls present, this accident sequence was deemed to be not credible.

XII. Effluent with High Uranium Concentration Transferred to EPB

Intervenors' brief (Intervenors' Pres. at 31) cites a criticality accident in the Effluent Processing Building (EPB) where solution with a high uranium concentration is transferred to the EPB.

Excess Uranium-1 (ISA likelihood index -5)

Ammonium hydroxide is added to uranyl nitrate inside the OCB to cause precipitation of ammonium diuranate. The effluent from this process is transferred to the uranium recovery area, where it is pumped through cross-flow filters to remove any entrained ammonium diuranate solids, and then through an ion exchange system. The ion exchange system reduces the uranium concentration in the effluent to 1 ppm or less. The effluent is then pumped outside the OCB to the EPB and into receiving tank TK-50. In the EPB ammonium hydroxide is recovered in a series of large non-favorable geometry vessels. The recovered ammonium hydroxide is then pumped back to the OCB for use in the precipitation process.

The uranium concentration in the effluent transferred to the EPB is very low under normal conditions. This accident sequence is concerned with the inadvertent transfer of solution with a high uranium concentration to the EPB. The subcritical concentration limit for LEU is 283 gU/L, which is approximately 283,000 times more concentrated than the normal effluent discharged from the ion exchange columns. In fact, this concentration is orders of magnitude greater than what is typically in the effluent fed to the uranium recovery process. The initiating event as it is represented in the ISA is that effluent with more than 1 ppm U is transferred from the ion exchange system to the EPB.

This initiating event is assigned a frequency index of -1, corresponding to an event expected to occur during the facility lifetime. However, note that the real initiating even needed to lead to a possible criticality accident is discharge of effluent from the ion exchange columns with a uranium concentration greater than 283 g/L, or approximately 283,000 ppm. The frequency index associated with such a process upset is much less than -1, since it requires not only failures in the uranium recovery area, but failures upstream of the uranium recovery area to cause such high concentration solution to be fed to the uranium recovery area.

Two inline concentration monitors provide protection against transfer of effluent with a high uranium concentration to the EPB. Each monitor is a colorimetric analyzer that accurately detects uranium concentrations based on absorption spectrometry. Each monitor is interlocked to a three-way valve and to a block and bleed valve. Upon detection of effluent with a uranium concentration of 50 ppm or higher, the system causes the three-way valve to redirect effluent back to the ion exchange system feed tank, instead of to the EPB feed tank (TK-50). It also causes a block and bleed valve arrangement on the transfer line to the EPB to close. This valve arrangement consists of two block valves with a bleed valve located between them. Upon activation, both of the block valves close and the bleed valve opens. If any solution leaks past the first bleed valve it drains out the bleed valve. This is an extremely effective barrier.

Both of the colorimetric analyzers are designated as IROFS, and each is conservatively assigned a failure frequency index of -2. Clearly a -3 index could be justified for these robust active engineered controls, particularly in light of the fact that the set point is orders of magnitude lower than the minimum critical concentration.

A third colorimetric analyzer is located on the same transfer line as the first two and performs in a similar manner. However, it is not designated as an IROFS. Therefore, its failure is treated as an enabling event with a frequency index of -1. This is again clearly a highly conservative assignment.

To summarize, effluent with a concentration thousands of times higher than that normally fed to the uranium recovery system would have to be discharged from the uranium recovery system, and three colorimetric analyzers would have to fail, before a criticality could occur due to high uranium concentration in the EPB. This accident sequence probably is not credible. The index assignments in the ISA were very conservative, and the ISA did not credit other events that would have to occur before criticality would be possible.

XIII. Accumulation of Uranium in the Effluent Processing Building

Intervenors' brief (Intervenors' Pres. at 31) cites a criticality accident in the Effluent Processing Building (EPB) where uranium accumulates over time in equipment in the EPB.

Excess Uranium-2 (ISA likelihood index -5)

This accident sequence was assigned a -5 index in Rev. 0 of the NCSE, which was reported in the ISA reviewed by Intervenors. However, the sequence was subsequently declared not credible in Rev. 1 of the NCSE, due to a better understanding of the process. The discussion below is based on Rev. 1 of the NCSE and shows that the not credible designation is warranted.

As discussed above in Section XIII, the uranium concentration in the effluent entering the EPB is normally 1 ppm or less. The minimum uranium critical mass given the chemical form of uranium present in the EPB is 32.6 kg. Considering the effluent flow rate into the EPB, it would take 5 years before the cumulative mass of uranium flowing into the EPB would reach the minimum critical mass. Further, for criticality to be possible, all of the uranium in the solution entering the EPB over a five year period would have to accumulate in a single vessel. Thus, this accident is not possible at normal effluent concentrations. Recall that the colorimetric analyzers have a set point at 50 ppm. If some initiating event occurred in the uranium recovery process such that effluent with a uranium concentration of 50 ppm were transferred to the EPB for an extended period of time, then a critical mass could hypothetically accumulate in 38 days. However, this also would require that all of the uranium accumulate in a single vessel. The largest vessel in the EPB is the receipt tank, which has a 4,000 gallon volume. At a nominal flow rate this tank is filled every 27 hours. Therefore, any significant uranium accumulations in the EPB would have to be distributed in equipment throughout the system rather than all being accumulated in a single vessel. Thus, even if every bit of the uranium entering the facility remained as holdup, and even if an extenuated initiating event caused continuous transfer of 50 ppm effluent to the EPB, substantially more than 38 days would be required before a critical mass could accumulate anywhere. Moreover, even then, for criticality to result, the uranium would have to somehow accumulate in a geometry that would enable it.

A mass balance is performed on the EPB on a monthly basis. Grab samples are automatically taken from the transfer line from the ion exchange system to the EPB receipt tank. The composite sample representative of a week is analyzed in the laboratory to determine the uranium mass that entered the EPB during that time. For the same time period, laboratory analysis of the solid and liquid wastes establishes the uranium mass exiting the EPB. The difference between the entering and exiting mass is the uranium holdup in the EPB. If the cumulative holdup in the EPB exceeds 14 kg UO_2, tank inspections and NDA scans of equipment are performed to locate areas of high holdup and clean them out. Since the mass balance is performed monthly, it is not possible for more than a critical mass to accumulate in any one vessel in between mass balances.

Clearly accumulation of a critical mass of uranium in a vessel in the EPB is not credible.

XIV. High Uranium Concentration in TK-10 Due to Evaporation of Spilled Solution

Intervenors' brief (Intervenors' Pres. at 31) cites a criticality accident in the UNB where spilled uranium solution is concentrated via evaporation and subsequently pumped into TK-10.

TK-10 High U Concentration (ISA likelihood index -5)

This accident sequence was assigned a -5 index in Rev. 0 of the NCSE, which was reported in the ISA reviewed by Intervenors. However, the sequence was subsequently declared not credible in Rev. 1 of the NCSE, due to a better understanding of the process. The discussion below is based on Rev. 1 of the NCSE and shows that the not credible designation is warranted.

The UNB is equipped with a spill basin where spilled solution will accumulate. Solution in the spill basin can be pumped into TK-10. This accident sequence posits a spill that goes undetected for a long enough period of time that evaporation causes the spilled solution to concentrate to a uranium concentration that exceeds the safe subcritical value of 283 g/L. This concentrated solution is then pumped into TK-10 where the criticality occurs.

This accident sequence is initiated with the occurrence of a large spill. Calculations show that the worst case occurs if the volume of spilled uranyl nitrate that accumulates in the spill basin is approximately 1,000 gallons. Spill volumes more or less than this amount result in a lower surface area to volume ratio, and therefore are less efficient for evaporation. If the spill volume consists of 1,000 gallons of uranyl nitrate solution at an initial concentration of 231 g/L, evaporation could cause the concentration to increase to 238 g/L after 23 days (assuming optimal conditions – in winter months such evaporation obviously would not occur). It is important to note that the starting point for the calculation was 231 g/L, which is the Limiting Condition for Operation (LCO) for the UNB. The Routine Operating Limit (ROL) is 210 g/L, and solution in the UNB normally has a concentration between 125 g/L and 210 g/L. Obviously if the concentration is lower the time required to reach 283 g/L via evaporation is extended.

The major reason this accident sequence is not credible is because operations personnel walk through the facility numerous times per day, and security personnel walk-down the facility at least once per day. A large spill would be very visible and obvious to anyone who enters the facility. Therefore, a large volume spill would be detected and cleanup procedures initiated very soon, probably within one day of the spill occurring. In addition to visual detection, the facility is equipped with leak detectors that alarm in the control room.

When a spill is detected procedures require operators to identify its source before proceeding. If this cannot be accomplished, samples are taken to determine the uranium concentration in the solution. If the concentration exceeds 210 g/L, deionized water is used to dilute it before pumping it into TK-10.

The recirculation line on TK-10 is equipped with a density monitor interlocked to a valve on the tank inlet line. If the measured uranium concentration exceeds 210 g/L, the system causes the valve to close, thus terminating any transfers.

Spills will occasionally occur in the UNB, and the spilled solution will be pumped into tank TK-10. However, it is not credible that the spilled solution would concentrate via evaporation to a value greater than the minimum value required for criticality.

XV. High Volume/Enrichment in the Staging Columns

Intervenors' brief (Intervenors' Pres. at 30) cites criticality accidents in the BPF Downblending process where too much HEU solution is transferred into the blend tank, or the enrichment of the HEU transferred to the blend tank is too high.

High Volume in Staging Columns (ISA likelihood index -5)
High Enrichment in Staging Columns (ISA likelihood index -5)

It should be noted that the version of the NCSE that supported the ISA version reviewed by Intervenors reported -5 risk indices for both of these accident scenarios. The current revision of the NCSE (Rev. 3) defends a -6 risk index for each of the sequences. The discussion below demonstrates that this assignment is justified, and that there are additional conservatisms even beyond the -6 assignment.

The purpose of the downblending system is to add HEU solution to a solution of naturally enriched uranium and produce an LEU product. The downblending tank itself is of favorable geometry for LEU but not for higher enrichments. Therefore, it is very important that the enrichment of the solution in the downblending tank be controlled. The parameter that is actually controlled is the concentration of U-235, which is limited to 11.78 g/L.

A blend batch is initiated by adding naturally enriched uranium solution to the blend tank. A minimum volume of approximately 13,500 liters is required. To assure this volume is added and always present in the tank, two independent level indicators are installed on the blend tank. These are interlocked to the pump that must be running in order for the transfer of the HEU solution to occur. Therefore, the transfer of HEU solution cannot begin unless at least 13,500 liters of naturally enriched solution is present in the blend tank, and the transfer is terminated if at any time during the blend process the volume decreases below this limit.

HEU solution is prepared in a set of favorable geometry columns referred to as the mix and measure columns. Two HEU streams of differing enrichments and concentrations are combined in the mix and measure columns to achieve a feed with the desired enrichment and concentration, but the maximum permitted values of each are 65% and 365 g/L. Once the desired solution parameters are achieved the solution is transferred to another set of favorable geometry columns referred to as the staging columns. The volume of solution in the staging columns is also determined by the blend recipe such that the final blended solution will have the desired characteristics.

The blending batch is initiated by starting a pump that recirculates the naturally enriched solution in the blend tank. A line from the staging columns taps into this recirculation line in manner that creates a Venturi effect, such that the HEU solution is sucked from the staging columns into the recirculation pipe. The HEU solution mixes with the natural solution in the pipe and then enters the blend tank.

There are several ways in which the LCO ^{235}U concentration of 11.78 g/L could be exceeded. Two of them are the subject of the accident sequences quoted by Intervenors: a high volume of HEU is transferred from the staging columns to the blend tank, and high enrichment HEU solution in the staging columns is transferred to the blend tank. Each of these accident scenarios is discussed below.

More than 600 liters of HEU solution would have to be transferred from the staging columns to the blend tank before the LCO could be exceeded, assuming the HEU concentration and enrichment did not exceed the maximum allowable values, and the minimum required volume of naturally enriched uranium was present in the blend tank. Volume measurements are initially taken in the mix and measure columns, which are filled to the volume required by the blend recipe (which is always ≤ 600 liters), using precision instrumentation. The initiating event for the accident sequence then is accumulation of more than 600 liters in the mix and measure columns, either due to valve failure or level indicator failure, or a combination of the two. This initiating event is assigned a frequency index of -1, corresponding to an event expected to occur within the facility lifetime. The HEU is then transferred to the staging columns, where the operator is required to verify the volume against the blend recipe again using precision instrumentation. Failure to properly perform this step or failure of the level indicators on these columns is an enabling event that is also assigned a failure frequency index of -1.

Once the pump initiates the Venturi action, HEU solution begins to flow from the staging columns into the blend tank via the Venturi. A volumetric mass flow meter on the transfer line accumulates the total volume of solution that flows through the pipe and terminates the transfer (by closing two valves on the transfer line) if the volume exceeds 600 liters. This robust active engineered control is assigned a failure frequency index of -2, although a -3 index is probably justifiable.

In addition, an inline gamma monitor is mounted on the recirculation line and is calibrated to detect ^{235}U concentration. If the concentration exceeds the LCO value of 11.78 g/L, it causes the two valves on the transfer line to close. This is another robust active engineered control that is also assigned a failure index of -2, although again a -3 index is probably justifiable. The total likelihood index for the accident scenario is -6, although a -8 index is probably justifiable.

Even though the likelihood index of -6 is low, and the justifiable likelihood index of -8 is extremely low, there are conservative aspects not credited in the ISA. There is a very large margin of safety between the LCO at 11.78 g/L, and the license subcritical concentration limit of 13.86 g/L. Considering the tank minimum blend stock (natural U) volume of 13,500 liters, the mass required to increase concentration from 11.78 g/L to 13.86 g/L is 28 kg. At the maximum permitted HEU concentration of 365 g/L and the maximum permitted enrichment of 65%, that means 118 liters of HEU beyond the volume required to achieve 11.78 g/L would have to be added to the blend tank. That is a very large volume error, when one considers the maximum volume added in a blend is limited to 600 liters.

In summary, this accident sequence is demonstrated in the NCSE to have a likelihood index of -6, but a -8 index could be justified. Also, the extra volume of HEU that would have to be added to cause a criticality is very large relative to the total volume permitted to be added, which significantly reduces the probability that the sequence would occur.

The second related accident sequence referenced by Intervenors is the transfer of high enriched solution from the staging columns to the blend tank. Recall that the enrichment of the HEU solution added to the blend tank is not permitted to exceed 65%. The enrichment of the HEU solution is determined via dual sampling at three locations. As discussed earlier, there are two sources of HEU solution to the mix and measure columns. Dual sampling is performed at both of these sources prior to transfer to the mix and measure columns. Once in the mix and measure columns, a final enrichment check is accomplished via dual sampling in those columns. In all cases dual sampling means two samples are drawn from the columns and then independently analyzed in the laboratory. If the results of the two measurements are not close then a third analysis is performed, or the system may be resampled. Therefore, the dual sampling routine is an extremely robust administrative control and it is assigned a -2 failure index both at the upstream sample locations and at the mix and measure column location. Recall that an inline concentration monitor is mounted on the recirculation line of the blend tank and that it terminates the transfer from the staging columns if the ^{235}U concentration exceeds the LCO. This active engineered control is again assigned a failure frequency index of -2, although a -3 index is probably justifiable. The total likelihood index for this accident sequence is -6, although a -7 could be justified.

XVI. Conclusion

The foregoing discussions demonstrate the inherent conservative nature of the ISA criticality accident evaluations, in that in every accident sequence/scenario cited by intervenors (and indeed in all accident sequences/scenarios) there are several unlikely events that would have to occur beyond those credited in the ISA before a criticality would be possible. NRC safety regulations require that all credible criticality accident sequences be rendered highly unlikely through the utilization of safety systems (IROFS). Thus, in fact, the additional unlikely but uncredited events in the accident sequences render a criticality accident even more unlikely. Therefore, we have shown that the intervenors have misestimated and significantly overestimated the likelihood of criticality accidents associated with the BLEU Project.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 14, 2004.

Robert L. Frost

Robert L. Frost

Qualifications of Robert L. Frost

Robert L. Frost is the President of Nuclear Safety Associates, a nuclear safety services company with headquarters in Johnson City, TN. He holds a B.S. in Chemical Engineering and a Ph.D. in Nuclear Engineering, with minors in electrical engineering, mathematics, and chemistry, earned from the Georgia Institute of Technology. He began his career performing reactor physics safety analyses for the isotope production reactors at the Savannah River Site as an employee of Westinghouse Savannah River Company (WSRC). Within several years Dr. Frost had expanded his fields of expertise to include radiation shielding and nuclear criticality safety. He mastered a large number of highly complex computer codes, including those that use both Monte Carlo and discrete ordinates methodologies to solve the Boltzmann Transport Equation for 1, 2, and 3D geometries. In 1991, less than one year out of school, Dr. Frost was appointed the leader for a team of engineers performing the safety analysis for the Savannah River Site's (SRS's) K-Reactor. He implemented several improvements in analyses methodologies that allowed the team to complete their portion of the Safety Analysis Report on time and within budget. Dr. Frost went on to lead several other team efforts at SRS, including the criticality safety and shielding efforts to support Chapters 5 and 6 of the Safety Analysis Report for Packages (SARP) for a variety of shipping packages. These included the 9972 and 9975, which were successfully licensed and are extensively utilized at several Department of Energy (DOE) sites, and the 5320, which also was successfully licensed and whose function is to transport Pu-238 that is used in radioisotope power systems for deep space missions.

Dr. Frost was twice awarded the George Westinghouse Signature Award of Merit while at SRS. The first time for his leadership in developing process improvements to the reactor safety analysis methodology, including changes to computer codes. The second award was for technical innovation in solving a long-standing problem at the SRS. For decades the site was aware of an inability to accurately model the axial neutron flux in the site's production reactors. Through research and implementation of novel computational techniques Dr. Frost was able to solve this problem. This ultimately led to an increase in the regulator's confidence in the reactor physics computer codes utilized on site.

Another of Dr. Frost's accomplishment's at SRS was the authorship of RASTA (Radiation Source Term Analysis), a complex computer code used for determining the radiation emanating from various materials, accounting for gamma, beta, and alpha radiation from isotopic decay, production of neutrons from (α,n) reactions, and neutron and gamma radiation from spontaneous fission. He wrote the code because existing codes were inadequate for analysis of the radiation source term from a tank at SRS that contained huge quantities of higher actinides (e.g., Pu, Np, Am, Cm, etc) in solution. The results of this work were presented at a national meeting of the American Nuclear Society and garnered a lot of attention due to the unique nature of the material.

Dr. Frost's mentor in the field of nuclear criticality safety was Mr. Jim Mincey, who was very well known and respected in the industry (he passed away in 2002). Mr. Mincey instilled in Dr. Frost the importance of chasing down every detail, of expecting the unexpected, and how to unearth accident sequences that others might miss. His guidance gave Dr. Frost a firm foundation upon which to build his criticality safety skills.

In 1995 Dr. Frost traveled to Carlsbad, New Mexico, to learn about the operations at the Waste Isolation Pilot Plant (WIPP). This facility is now open and serves as the nation's only repository for transuranic (TRU) waste. It consists of tunnels and rooms dug out of thick salt layers more than 2,000 feet below the surface. Dr. Frost wrote the Nuclear Criticality Safety Evaluation (NCSE) to determine the conditions under which the site could operate without concern of an inadvertent criticality event. This required extensive research and a detailed understanding of the site environment during both the operational and post-closure stages. After the site is closed and maintenance is ceased, the salt walls will cave in on the rooms and tunnels, crushing the drums remaining inside. This unique phenomenon had to be accounted for in the criticality analysis. The NCSE Dr. Frost wrote was an integral part of the safety basis on which permission was granted for the site to open, and it still serves as the basis for criticality safety at the WIPP.

In 1997 Dr. Frost spent 6 months supporting the criticality safety group at the Department of Energy's (DOE's) Y-12 plant in Oak Ridge, Tennessee. There he quickly established a reputation for going "outside the box" in finding innovative solutions to longstanding problems. His most significant contribution was in successfully addressing a concern raised by the Defense Nuclear Facility Safety Board (DNFSB) regarding the criticality implications of activation of the fire sprinkler system in areas where large quantities of HEU are stored and/or processed. Dr. Frost also developed criteria for use in identifying, isolating, and removing HEU deposits in ductwork at the facility, and supported daily operations.

Dr. Frost began supporting the criticality safety group at Nuclear Fuel Services (NFS) in late 1997, working as a contractor in a staff augmentation role. He initially worked to support all uranium recovery operations (solvent extraction, dissolution, calcination, precipitation, evaporation, etc) in the Navy Fuel Manufacturing Facility. He is ideally suited for such work due to his chemical engineering background. He also served as a technical reviewer for many of the fuel production operations, and as a mentor for the younger engineers. Within two years he became recognized as the senior nuclear criticality safety (NCS) expert at NFS.

Dr. Frost has also worked extensively with NCS implications of waste water operations associated with HEU facilities. He has presented several papers on this subject at American Nuclear Society meetings, including one discussing his novel application of an inline gamma detector for controlling uranium mass in large, non-favorable geometry tanks. Other papers have covered diverse topics such as the implications of uranium precipitation and chemistry control in waste water operations, and the reactivity effects associated with uniformly-distributed small uranium particles.

NFS completely rebuilt its HEU vault in 2002. Dr. Frost performed an exhaustive study of storage options for a wide array of material types and worked closely with the engineering designers. The result is a state-of-the-art facility that relies heavily on passive-engineered safety features. The design was praised by NRC inspectors due to the extensive use of these passive features and because of the exhaustive nature of the analysis, which inspires great confidence in the safety of the operation.

In 1999/2000 Dr. Frost served as the criticality safety representative on a team of engineers contracted to design and license the Trans Nuclear FSV shipping cask to carry a large variety of research and test reactor fuels stored at the Oak Ridge National Laboratory (ORNL). This cask was a one-of-a-kind due to the unique nature of the cargo. Dr. Frost performed a highly-sophisticated analysis to optimize the loading patterns for each shipment, interfacing with the shipping agency. He worked closely with the engineering group to find a design that would work. When the Safety Analysis Report for Packages (SARP) was submitted to the NRC for its licensing review, Dr. Frost's chapter (Chapter 6) was accepted without comment, an unheard of accomplishment in the shipping package licensing arena. Dr. Frost presented a paper on this effort at a national meeting of the American Nuclear Society.

Dr. Frost is an active member of the Nuclear Criticality Safety Division (NCSD) of the American Nuclear Society. He currently serves as Chairman of the Program Committee, where his responsibilities include determining the subjects for paper sessions at the national meetings, reviewing and having ultimate authority for accepting or rejecting submitted papers, and assigning session chairpersons. He also is a member of the Division's Executive Committee, where he participates in the general governance of the Division.

Dr. Frost formed Nuclear Safety Associates (NSA) in April of 2001. NSA is a nuclear safety services company that maintains expertise in various fields of nuclear safety, with a core strength in the field of nuclear criticality safety. NSA was founded with the concept of adding only staff members with a high degree of competence and who are respected among their peer group, and the current 13 staff members are representative of this commitment to quality. NSA has over ten clients, including NFS.

NuclearSafety
Associates

Robert L. Frost, Ph.D.
Senior NCS Specialist

AREAS OF EXPERTISE:

- *Nuclear Criticality Safety*
- *Radiation Shielding*
- *Packaging and Transportation - SARP Development*
- *Nuclear Reactor Safety Analysis (Neutronics)*

SUMMARY:

Dr. Frost is a Senior Nuclear Criticality Safety and Shielding Specialist. He has over 16 years experience in the nuclear safety field, and 13 years of experience performing and reviewing NCS and Radiation Shielding calculations and analyses. Within these fields, he has extensive experience in both facility support and the sub-specialty of Packaging and Transportation. Dr. Frost has authored NCSE's and supported daily operations for all types of uranium recovery operations, vault storage facilities, waste-water operations, fissile holdup in ductwork, and a variety of operations related to the Navy fuel production facility. Additional areas of expertise include validation and benchmarking of criticality codes and reactor physics calculations in support of SAR efforts. He has also played a leading role in the evolution of the criticality safety program at NFS. Dr. Frost is an experienced user of KENO V.a, KENO VI, MCNP, and MORSE (Monte Carlo neutron transport), DANTSYS, DORT/TORT and associated utilities (discrete ordinates neutron transport), QAD-CGGP (point-kernel), ORIGEN-S (depletion and source term analysis), the SCALE system (including SAS and CSAS sequences), and the AMPX-77 System (cross section processing). Dr. Frost is the developer of the RASTA code for source term analysis and numerous utilities. He is proficient in DOS, Windows, UNIX, FORTRAN, and HTML. Dr. Frost is an active member of the Nuclear Criticality Safety Division Program Committee, and also participates on the executive committee.

EDUCATION:

- Ph.D., Nuclear Engineering, Georgia Institute of Technology, 1990
  Minors in Electrical Engineering and Mathematics
- B.S., Chemical Engineering, University of South Florida, 1986
  Minor in Chemistry

CLEARANCES: DOE Q (Active), NRC U (Active)

EMPLOYMENT HISTORY:

Nuclear Safety Associates, Inc. — *April 2001 to present*
President & Senior Consultant

Navarro Research & Engineering, Inc. — *Nov 1997 to April 2001*
Senior Criticality Safety Specialist

Westinghouse Savannah River Company, Inc. — *Sept 1990 to Nov 1997*
Senior Engineer A

Georgia Tech Research Center — *Sept 1987 to Sept 1990*
Graduate Research Assistant

PROFESSIONAL AFFILIATIONS: Member, American Nuclear Society

To discuss availability of this consultant,
please contact NuclearSafety Associates at (423) 610-0249

NuclearSafety Associates

Robert L. Frost, Ph.D.
Senior NCS Specialist

EXPERIENCE:

Client: Nuclear Fuel Services, Inc.
01/97 to present

Criticality safety support for the Navy nuclear fuel production facility and the uranium recovery operations. Recognized expert for uranium recovery operations, waste water processing, and vault storage. Developed a unique method of controlling mass in large tanks using in-line monitoring. Represent the criticality group on design teams, urging the use of passive or engineered controls when practical, and assuring that any administrative controls necessary will be simple and effective. Participate on multi-disciplinary panel to develop the Hazard Analysis, using a "what if" approach. Perform double contingency analyses conforming to the requirements of 10 CFR 70. Formally document results in Nuclear Criticality Safety Evaluations that are directly reviewed and approved by the NRC. Provide daily floor support of operations and training of operators.

Client: Trans Nuclear Inc., Bechtel Jacobs
11/99 to 12/00

Performed sophisticated criticality safety analyses for the Oak Ridge Container in the Trans Nuclear FSV Shipping Cask. Performed all calculations required to demonstrate compliance with 10 CFR 71 criticality requirements, and wrote Chapter 6 of the SARP. **Review by NRC produced no criticality RAI questions.**

Client: Lockheed Martin Energy Systems/Y-12 Plant
06/97 to 01/98

Supported restart of Enriched Uranium Operations at the Oak Ridge Y-12 Plant, primarily the uranium recovery operations in Buildings 9212 and 9215. Alleviated a DNFSB concern regarding criticality safety during fire sprinkler activation. Developed requirements for ductwork cleanout. Supported daily operations in Building 9215, performed audits, etc.

Client: The Savannah River Site
09/90 to 09/97

As Team Leader for criticality and shielding support of packaging and transportation tasks, performed criticality analyses and directed work of other team members, wrote Chapter 6 of the SARP, and responded to DOE review comments for the 9972 family of drum packages, the 5320 plutonium oxide package, the LR-56 plutonium solution transportation system, and several packages for on-site transport. Performed the shielding calculations and wrote Chapter 5 (Shielding) of the SARP for the 5320, LR-56, and several on-site packages. Performed the criticality analysis and wrote the NCSE for contact handled waste in the Waste Isolation Pilot Plant (WIPP). Performed shielding evaluations to demonstrate compliance with personnel exposure limits for several Savannah River Site fuel reprocessing operations, including the Am/Cm processing facility in F-Canyon. Developed the RASTA code package for photon and neutron radiation source term development, which is used extensively by Westinghouse Safety Management Solutions to create radiation source terms for shielding studies. Participated in the effort to recover the LLNL plutonium button experiments for the International Handbook of Evaluated Criticality Safety Benchmark Experiments (PU-MET-FAST-004). Supported the SRS K-Reactor restart effort. Using unique algorithms, solved a long-standing problem in predicting core axial flux shapes. Awarded the George Westinghouse Signature Award of Merit in 1994 for this effort. Participated on a team that developed an automated system to set up 3-D diffusion theory models of the SRS reactors. The team was awarded the George Westinghouse Signature Award of Merit in 1993 for this work.

PUBLICATIONS

Robert L. Frost, Controlling Fissile Mass or Concentration in Large Tanks Using an Inline Monitor, Proceedings of the Embedded Topical Meeting on Practical Implementation of Criticality Safety, November 11-15, 2001, Reno, NV, USA.

R.L. Frost, R.L. Webb, and S. Kahook, *Comparison of TORT and MCNP for Thermal Flux Calculation in a Long One-Bend Corridor Model*, Proceedings of the 1998 ANS Radiation Protection and Shielding Division Topical Conference, April 19-23, 1998, Nashville, TN, Volume 1, pp. I-132 - I-140.

R.L. Frost, *RASTA: A Generalized Tool for Radiation Source Term Analysis*, Transactions of the American Nuclear Society, 77 1997, p312.

R.L. Frost, *Re-Evaluation of Three Plutonium/Oralloy Composite Systems for Use as Critical Benchmarks*, Proceedings ANS 1994 Winter Meeting, Nov. 13-17, 1994, Washington, D.C.

K.A. Niemer, **R. L. Frost**, and T.G. Williamson, *Equivalence Relations for Mixtures of Nuclides in Savannah River Site Shipping Casks*, Proceedings of DOE Topical Meeting on Spent Nuclear Fuel, Dec. 10-17, 1994, Salt Lake City, UT.

E. F. Trumble , J.B. Justice, and **R.L. Frost,** *Lawrence Livermore Plutonium Button Critical Experiment Benchmark*, Proceedings of the 1994 ANS International Meeting, June 19-23, 1994, New Orleans, LA.

R.L. Frost, *A New Axial Geometry Model for K Reactor*, ANS Transactions 58A (436) 1993.

D.L. George and **R.L. Frost,** *A Study of Temperature Coefficients of Reactivity for a Savannah River Site Tritium-Producing Charge (U),* Proceedings of the 1992 Topical Meeting on Advances in Reactor Physics, March 8-11, 1992, Charleston, SC.

R.L. Frost, A.B. DeWald, A. Rohatgi, M. Zaluzec, J.M. Rigsbee, B. Nielsen, and K.G. Lynn, *Slow Positron Annihilation Spectroscopy and Electron Microscopy of Cobalt and Nickel Silicide Thin Films*, J. Vac. Sci. Technol. A. (July/August, 1990).

K.V. Logan, W.L. Ohlinger, J.T. Sparrow, **R.L. Frost**, F. Saterlie, and J. Fryer, *Behavior of New Antenna Window Materials During High Temperature Permitivity Measurements*, Proceedings of the 3rd DOD Electromagnetic Window Symposium, Nov. 1989, Redstone Arsenal, Huntsville, AL.

J.P. Schaffer, A. Rohatgi, A.B. DeWald, and **R.L. Frost,** *Defect Characterization in Semiconductors by Positron Annihilation Spectroscopy*, J. Electronic Mat. 18, 737 (1989).

J.P. Schaffer, A.B. DeWald, **R.L. Frost,** A.J. Perry, B. Nielsen, and K.G. Lynn, *Positron Annihilation Spectroscopy of the Defect Structure of Sputter Deposited TiN*, Surf. Coat. Tech. 36, 593 (1988).

R.L. Frost, A.B. DeWald, J.P. Schaffer, A. Rohatgi, B. Nielsen, and K.G. Lynn, *Slow Positron Annihilation*

Spectroscopy of Hetero and Homo Junctions of GaAs Based Semiconductor Thin Films, Thin Solid Films 166, 349 (1988).

A.B. DeWald, **R.L. Frost**, S.A. Ringel, J.P. Schaffer, A. Rohatgi, B. Nielsen, And K.G. Lynn, *Positron Annihilation Spectroscopy of AlGaAs/GaAs interfaces in MOCVD-Grown GaAs Heterojunction Solar Cells*, J. Vac. Sci. Technol. A6, 2248 (1988).

ATTACHMENT 3

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Presiding Officer

In the Matter of	)	
	)	Docket No. 70-143
Nuclear Fuel Services, Inc.	)	Special Nuclear Material
	)	License No. SNM-124
(Blended Low Enriched Uranium Project)	)	

Declaration of John R. Frazier Regarding the Dispersion of Airborne Effluents

John R. Frazier states as follows under penalty of perjury:

I. INTRODUCTION

My name is John R. Frazier, Ph.D., and I am an Associate with Auxier & Associates, Inc., in Knoxville, Tennessee. The purpose of this declaration is to respond in part to the October 14, 2004 written presentation by the State of Franklin Group of the Sierra Club, Friends of the Nolichucky River Valley, Oak Ridge Environmental Peace Alliance, and Tennessee Environmental Council (henceforth referred to as "Intervenors") on behalf of Nuclear Fuel Services, Inc. (NFS), in support of NFS's applications for license amendments authorizing operations associated with the Blended Low Enriched Uranium ("BLEU") Project. This declaration pertains to atmospheric dispersion of airborne releases from the BLEU project into the valley in which the NFS site is located and radiation doses from routine and accidental releases that might occur from the BLEU Project. The conclusions presented in this declaration are in accordance with the concepts, methodologies, and procedures that are generally accepted in the fields of health physics and radiation safety.

II. QUALIFICATIONS

My area of expertise is health physics and radiation safety. My academic degrees include a B.A. in physics, an M.S. in physics, and a Ph.D. in physics, with emphasis in health physics. I have over 27 years of professional experience in health physics, primarily in the areas of environmental and occupational radiation dose assessments, external radiation dosimetry, internal dosimetry, radiobioassay, radiation dose reconstructions, radiation protection standards and regulations, radiation detection and measurement, and collection and interpretation of environmental characterization data. I earned Comprehensive Certification by the American Board of Health Physics in 1981 and have been recertified every four years since that date, with the most recent recertification through 2005. I am a Fellow and Past President of the Health Physics Society and a Diplomate of the American Academy of Health Physics. I am an elected member of the National Council on Radiation Protection and Measurements. My further qualifications and experience as a health physicist are detailed in my Curriculum Vitae (Attachment 1).

The Environmental Protection Agency (EPA), the U.S. Department of Agriculture (USDA), the U.S. Department of Defense (DOD), the U.S. Department of Energy (DOE), the Defense Nuclear Facilities Safety Board (DNFSB), the U.S. Department of Justice (DOJ), and the Peace Corps of the United States have sought my advice on a wide range of health physics and radiation protection topics from environmental radiation dose and risk assessments to operational health physics program design. I have also served as a consultant to private companies and individuals on health physics and radiation safety issues.

I am familiar with the terms and concepts of health physics and radiation protection, having studied these topics for more than 35 years. I have personally performed and supervised evaluations and assessments of actual and postulated radiological conditions at numerous sites, including detailed characterization of radiation levels and radioactivity levels at nuclear facilities and in their surrounding environments. I am familiar with the concepts, models, and procedures for assessing radiation doses from external radiation sources and from internally-deposited radionuclides. I have personally performed exposure pathways analyses, radiation dose assessments, and dose reconstructions for numerous individuals at many sites in the U.S. I have taught many hundreds of persons the basic principles, concepts, and terminology of the science of health physics, radiation safety, and environmental radiation dose assessment and dose reconstruction.

III. ISSUES

A. Unconventional Dispersion or "Trapping" of Airborne Effluents

Based on site-specific meteorological data, materials (radioactive or non-radioactive) potentially released into the air from NFS operations associated with the BLEU Project (under normal operating conditions or in the event of any of the accidents that have been postulated) would be dispersed according to conventional atmospheric dispersion models and parameters and would not exhibit unconventional dispersion as a consequence of the location of the NFS site in a valley. "Trapping" of airborne releases in or near the NFS site is inconsistent with representative, site-specific meteorological data that show that adequate and sustained atmospheric dispersion conditions are present at the NFS site throughout the year.

In their October 14, 2004 written presentation, the Intervenors made reference (at the pages indicated) to the valley in which the NFS site is located. The specific statements by the Intervenors are:

> "... the location of the plant in a narrow mountain valley that may trap accidental airborne releases of chemical and radiological contaminants, ..." (page 3)
>
> "In this case, the unique characteristics of the Erwin area – including its high population, narrow valley geography, and riverside location – call for the preparation of an EIS to ensure that the impacts of the proposed BLEU Project on the local environment are thoroughly considered." (page 37)
>
> "There is no attempt to relate the effects of accidents to the particular environment surrounding the NFS-Erwin plant, including the community around the plant, the narrow mountain valley, or the Nolichucky River." (page 37)
>
> "Finally, preparation of an EIS is compelled by the unique characteristics of the surrounding environment, including the close proximity of a large and vulnerable population, the site's location in a steep and narrow mountain valley, and the proximity of a river that is a valuable recreational and economic resource as well as a drinking water source." (page 40)

The Intervenors do not give specific consequences of NFS being located in a mountain valley, but their statements imply that airborne releases from the BLEU Project would produce inordinately high concentrations of airborne materials (radioactive and/or non-radioactive) that would be confined to the valley, leading to significant exposures of offsite members of the public. Site-specific meteorological data acquired by the NFS over a representative, five-year period (1991-1995) show that the wind patterns within the valley occur from all directions with few periods of calm and that the average wind speed is approximately 3 meters per second (m/s). These meteorology data were presented in the December 1996 NFS Environmental Report (Section 3.2) and, as a five-year average, are considered representative of current conditions. Such meteorological conditions do not lead to "trapping" of airborne materials in the valley, but rather such conditions help to ensure efficient mixing and dispersion of airborne materials throughout the year. In addition, the valley in which the NFS site is located is aligned with the prevailing wind patterns for the region (southwest to northeast), also leading to efficient atmospheric dispersion (mixing) of materials from the site. Thus, the Intervenors' statements suggesting that the location of NFS in a valley would trap airborne effluents and somehow increase the consequences of an accident at NFS are incorrect.

B. Modeling of the Dispersion of Airborne Contaminants after an Accident

Calculations of airborne concentrations of materials (radioactive and non-radioactive) in offsite locations due to specific accident sequences have been performed as part of the Integrated Safety Analyses (ISAs) for the BLEU Project license amendments. The assumptions that NFS made regarding atmospheric dispersion of materials released during postulated accidents involving the BLEU Project are conservative. In other words, the calculated concentrations of airborne materials at offsite locations are greater than the concentrations that would most likely occur during a potential accident. Thus, the potential exposure of offsite individuals to airborne materials in offsite areas has been overestimated.

For example, the wind speed assumed to be present during the postulated accident scenarios is 1 m/s (Hotspot Versions 8.03 [DOS] and 2.05), whereas the site-specific meteorological data show that the average wind speed at the NFS site is approximately 3 m/s (December 1996 NFS Environmental Report at Section 3.2). A greater wind speed leads to greater dispersion of released materials (thereby lowering the airborne concentrations at any location) and to more rapid transport of airborne materials over any potentially exposed offsite individuals (thereby lowering the exposure duration). The atmospheric stability class (condition) assumed to exist during postulated accident scenarios is assumed to be the most extreme stability class for stack releases and for ground surface releases (Class A and Class F, respectively), although the meteorological conditions necessary for these stability classes seldom occur at the NFS site. Classes B, C, D, and E are the atmospheric stability classes that correspond to the specific meteorological conditions at the NFS site (e.g., NUREG/CR-3332 at Chapter 2). Each of these leads to lower average offsite concentrations from a stack release and from a surface release than the classes assumed in the ISA calculations (e.g., NUREG/CR-3332 at Chapter 2). Lower offsite concentrations lead to lesser impacts from releases. In addition, it was assumed for each potential offsite exposure scenario in the ISAs that the exposed individuals were unprotected by buildings or other structures (i.e., outdoors) for the duration of each release and that they stood at the location of the highest calculated concentration of released material, with no consideration as to whether there is no residence at that location (Hotspot Versions 8.03 [DOS] and 2.05 and Appendices of Rev 0 of the Environmental Radiological Consequence calculations). The dispersion modeling also assumes that exposed individuals are always down wind of the release. Each of these assumptions leads to an overestimate of potential exposures to offsite personnel.

Calculations of potential exposures included in the ISAs are intended to present conservative, upper-bound exposures and this has been done. Use of atmospheric dispersion calculation parameters and offsite exposure assumptions that are consistent with the site-specific data for the NFS site and surrounding locations would lead to much lower offsite exposures than the exposures calculated in the subject ISAs.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 15, 2004.

John R. Frazier

JOHN R. FRAZIER, Ph.D., CHP

Professional Qualifications

Dr. Frazier has over 27 years of health physics experience in external and internal dosimetry, environmental dose assessment, radiation risk assessment, radiation spectroscopy, health physics training, bioassay, radiation detection and measurement, and radiological site characterization. Numerous federal agencies including the Nuclear Regulatory Commission (NRC), Environmental Protection Agency (EPA), U.S. Department of Agriculture (USDA), U.S. Department of Defense (DOD), and U.S. Department of Justice (DOJ) have sought his advice on a wide range of health physics and radiation protection topics from operational health physics program design to environmental radiation dose and risk assessments. He has also served as a consultant to private companies and individuals on numerous health physics issues. He is an elected member of the National Council on Radiation Protection and Measurements (NCRP). Dr. Frazier has made presentations on introductory and advanced health physics and radiation protection topics for professional society meetings, student groups, and public interest forums. His publications are in the areas of fundamental interactions of radiation with matter, radiation detection instrumentation, radiological site assessments, and external and internal radiation dosimetry.

Education

Ph.D., Physics, University of Tennessee, Knoxville, Tennessee; 1978.

M.S., Physics, University of Tennessee, Knoxville, Tennessee; 1973.

B.A., Physics, Berea College, Berea, Kentucky; 1970.

Registrations/Certifications

Certification by the American Board of Health Physics in 1981; recertified through 2005.

Experience and Background

1993 - Present

Senior Radiological Scientist, Auxier & Associates, Inc., Knoxville, Tennessee.

Dr. Frazier serves as senior consultant on radiation protection issues for private companies and government agencies. He performs assessments of internal and external radiation exposures, environmental radiation doses and radiological risks from occupational and environmental exposures. He also performs evaluations and assessments of all aspects of operational health physics programs. Dr. Frazier serves as technical advisor to organizations that perform environmental radiological assessments and risk assessments and that provide occupational radiation protection services in government and industry.

1986 - 1993 ***Senior Radiological Scientist, Nuclear Sciences, IT Corporation, Knoxville, Tennessee.***
Dr. Frazier served as senior radiological scientist and technical manager of the health physics consulting group within IT. He was responsible for health physics professional services provided by IT for federal, state, and local agencies, contractors, and private companies. These services included development of all aspects of the health physics programs for nuclear facilities, technical assessments and evaluations of existing health physics programs, and environmental and occupational radiation dose assessments. He served as technical advisor and task manager for radiological aspects of remedial investigations and feasibility studies (RI/FSs). He also served as manager and technical director for specific projects in areas that included design and implementation of environmental monitoring and sampling programs, assessment of operational health physics programs, and radiation dose and risk assessments for occupational exposures and environmental releases. Previous responsibilities included serving as senior technical consultant for upgrading Environmental Health and Safety Programs at the Department of Energy Rocky Flats Plant, Oak Ridge National Laboratory, and the Oak Ridge Y-12 Plant.

1980 - 1986 ***Health Physicist, Oak Ridge Associated Universities, Oak Ridge, Tennessee.***
Dr. Frazier developed and coordinated Oak Ridge Associated Universities (ORAU) health physics training programs. He taught health physics and radiation protection courses for several hundred students each year at ORAU Professional Training Programs. He developed new lectures, laboratory exercises, and training materials for health physics training for the Nuclear Regulatory Commission, Department of Energy, and corporate clients. In addition to his training responsibilities, Dr. Frazier served as division health physicist for the Manpower Education, Research, and Training Division of ORAU. He served as technical consultant to federal and state agencies, other training institutions, and ORAU clientele on environmental, health and safety issues. He evaluated radiation measurement and radiation protection instrumentation equipment.

1978 - 1980 ***Chief Radiation Physics Section, Bureau of Radiological Health, Rockville, Maryland.***
Dr. Frazier supervised research and support activities of a staff of seven health physics professionals and technicians. He planned and implemented radiation research projects pertaining to ionizing radiation detection/ measurement. He

scheduled personnel requirements in accordance with the scope of such projects. He coordinated support for external radiation dosimetry by the Radiation Physics Section for all other branches in the Division of Electronic Products. He supervised and performed multi-point calibrations of radiation detection/ measurement instruments per month. Dr. Frazier also assisted in planning radiation dosimetric surveys of large numbers and types of ionizing radiation sources to reduce population exposure. He coordinated environmental radiation dosimetry for extended geographical areas using external radiation dosimeters.

1977-1980 *Research Physicist, Bureau of Radiological Health, Rockville, Maryland.* Dr. Frazier calibrated X-ray detection/measurement instruments. He maintained radiation calibration secondary standards traceable to the National Bureau of Standards. He evaluated new X-Ray detection/measurement instruments with radio-frequency fields under controlled environmental conditions and a wide range of ionizing radiation fields. He also developed external radiation dosimetry techniques with both active and passive dosimeters.

Awards/Activities

Fellow, Health Physics Society, 2000
Elda E. Anderson Award, Health Physics Society, 1988
Senior Technical Associate, IT Corporation, 1988
Distinguished Technical Associate, IT Corporation, 1990
National Council on Radiation Protection and Measurements (NCRP)
Council Member, 2002-2008
Scientific Committee 46, 1999-2004

Professional Affiliations

Health Physics Society
(Plenary Membership since 1981; President, 2002-3; President-Elect, 2001-2; Board of Directors, 1992-5; Treasurer-Elect, 1997-8; Treasurer, 1998-2000)
American Academy of Health Physics (Secretary, 1996-1997, Director, 1998)
East Tennessee Chapter of the Health Physics Society (Past President)
International Radiation Protection Association (Plenary Membership)

Publications

Dr. Frazier has prepared or contributed to over 100 reports and publications in the fields of health physics and environmental science.

List of Publications

Frazier, J. R., "Negative Ion Resonances in the Fluorobenzenes and Biphenyl" Ph.D. Dissertation, University of Tennessee, Knoxville, Tennessee, 1978.

Frazier, J. R., "Low-Energy Electron Interactions with Organic Molecules: Negative Ion States of Fluorobenzenes," Journal of Chemical Physics, Vol. 69, No. 3807, 1978.

Frazier, J. R., "Performances of X-ray Measurement Instruments in RF Fields," HEW Publication (FDA) 78-8065 Rockville, Maryland, 1978.

Frazier, J. R., "A Dosimetry System for Evaluating Chest X-Ray Exposures," HEW Publication (FDA) 79-I 107, 1979.

Film Badge Dosimetry in Atmospheric Nuclear Tests, National Academy Press, Washington, D.C., 1989.

ATTACHMENT 4

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Presiding Officer

In the Matter of	)	
	)	Docket No. 70-143
Nuclear Fuel Services, Inc.	)	Special Nuclear Material
	)	License No. SNM-124
(Blended Low Enriched Uranium Project)	)	

Declaration of Robert L. Frost and John R. Frazier Regarding Intervenor's Claims of Consequences From the Tokai-Mura, Japan Criticality Accident

Robert L. Frost and John R. Frazier state as follows under penalty of perjury:

Intervenors state in their written presentation that one of the most serious accident risks posed by the BLEU Project is that of a criticality accident. Intervenors' Pres. at 26. They note that under NRC safety regulations, 10 C.F.R. § 70.61, the NRC automatically considers the consequences of a criticality accident to be "high." They then asserts that "[t]he potential offsite impacts of criticality accidents are well-known as the result of the September 30, 1999 criticality accident at the Tokai-Mura facility in Japan."[1] Intervenors then go on to make several assertions about the accident's consequences and imply that such consequences could occur at NFS as a result of a criticality accident involving the BLEU Project. NFS responds to the Intervenors' claims here.

To begin with, NFS understands that because of the potential hazard to nearby workers, NRC regulations require criticality accidents to be considered "high" consequence events. However, it is important to remember that in the ISA process under which the safety of fuel cycle facility processes is evaluated, two different scales are considered when determining the consequence category to assign to an accident sequence. See 10 C.F.R. §§ 70.61(b) and (c). One scale considers the potential consequences to onsite workers; the other considers the potential consequences to members of the public and/or the environment. Criticality accidents are automatically deemed high consequence due to the potential for large radiation exposures to onsite workers; in particular, the workers within a few feet of the accident. See 10 C.F.R. § 70.61(b)(1). However, based on NFS's assessment of the potential off-site consequences of criticality accidents, potential doses to offsite members of the public are very small. If criticality accidents were categorized based only on the potential impact they would have on members of the public, then every criticality accident sequence identified in the ISA for the BLEU Project facilities would be categorized as a low consequence event.

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[1] Ibid. (citing Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, *NRC Review of the Tokai-Mura Criticality Accident* (April 2000) ("NRC Report"), appended as Attachment 1 to SECY-00-0085, Memorandum to the Commissioners from William D. Travers, Executive Director for Operations (April 12, 2000)).

Intervenors use the criticality accident at the JCO facility in Tokai-Mura, Japan, as an example to support their claim that potential criticality accidents at the BLEU facilities pose a serious risk to the environment and/or members of the public. A brief description of that criticality accident and response to the intervenors' claims follows.

The criticality accident occurred at the JCO Fuel Fabrication Plant on September 30, 1999. This is the only criticality accident worldwide to occur in a commercial fuel fabrication facility.

The JCO plant had a large-scale operation for converting LEU, in the form of UF_6 and with a ^{235}U enrichment not greater than 5%, into UO_2 for commercial light water reactor use. It also had one building licensed to handle uranium with enrichments up to 20% for small scale special projects. The criticality accident occurred during operations in this building, when operators poured uranyl nitrate at 18.8% enrichment into a non-favorable geometry vessel. Two of the operators were killed in the accident. The conditions that led to this accident were: (1) inadequate regulatory oversight; (2) lack of an appropriate safety culture; and (3) inadequate worker training and qualification. These conditions do not exist at NFS or at any fuel cycle facility in the United States. This is supported by the conclusion of the NRC Report:

> *Based on the review the staff determined that the current NRC oversight program at commercial U.S. nuclear fuel fabrication, conversion and enrichment facilities makes a similar accident unlikely, and no revisions to NRC's oversight program are needed as a result of the lessons learned.*

The JCO plant site is very small (~300 x 500 meters) and is situated in a densely populated inner-city location. The close proximity of offsite members of the public to the site assured that any dose they received from a radiological event, including criticality, would be higher than the dose that members of the public could receive at any of the US fuel fabrication facilities (e.g., NFS), where the sites are larger and the surrounding population less dense.

Intervenors acknowledge in their filing that the NRC review of the Tokai-Mura criticality accident, as documented in the NRC Report, concludes that "there was no significant impact on the health of the public nor the environment from radiation or the release of radioactive materials because the amount was so small ..." See Intervenors' Pres. at 27 (exposures were "insignificant" (citing NRC Report at 2)).

Despite this acknowledgement, Intervenors state over 400 people were exposed to radiation in excess of NRC standards for public exposure as a result of the Tokai-Mura criticality accident. Intervenors' Pres. at 27 (citing 10 C.F.R. §§ 20.1301, 20.1302). This statement is incorrect. Intervenors have confused annual exposure limits based on normal operations, applicable to all NRC-regulated facilities, with accident exposure limits applicable to fuel cycle facilities like NFS.

In 10 CFR 20.1301, the NRC promulgates dose limits for individual members of the public resulting from licensed operations (i.e., normal conditions) with a maximum of 0.1 rem per year. It is important to understand that this limit applies to normal operation of the facility, not upset conditions, as Intervenors mistakenly imply. There is also a provision in 10 CFR 20.1301(d) that gives NRC the right, at its discretion, to approve annual doses to members of the public from licensed operations to be as high as 0.5 rem per year.

In 10 CFR 70.61, NRC sets dose limits for offsite members of the public in defining the consequence category into which an accident sequence must be placed as part of the ISA process. Accident sequences that result in doses to members of the public greater than 5 rem but less than 25 rem are deemed to have "intermediate" consequences. 10 C.F.R. § 70.61(c)(2). Accident sequences that result in doses to members of the public greater than 25 rem are deemed to have "high" consequences. 10 C.F.R. § 70.61(b)(2). It follows that accident sequences that lead to doses to offsite members of the public that do not exceed 5 rem have low consequences.

Figure 7 in the NRC Report tabulates radiation doses received in the Tokai-Mura accident by six different groups of people. Four of these groups (approximately 227 people) consisted of JCO workers located onsite when the accident occurred, one group (7 people) consisted of non-JCO workers located near the site, and the last group (~207 people) consisted of off-site members of the public. The last group (members of the public) is most relevant for this discussion because intervenors claim that the Tokai-Mura accident is relevant to determining the radiation exposure that could result for off-site members of the public in the event of a criticality accident at NFS. Figure 7 gives a breakdown of doses received by these people. The vast majority of the people in this group (~180 people) received doses not exceeding 5 mSv (0.5 rem). Therefore, of the 207 members of the public who received measurable doses, ~180 received a dose less than the limit NRC is authorized to grant a licensee as a normal operating condition. The remaining (~27) members of the public received doses greater than 0.5 rem but not exceeding 2.5 rem. The upper limit for this dose range is significantly less than the intermediate consequence lower threshold, which is also the low consequence upper threshold. Therefore, the consequences of the Tokai-Mura criticality accident, gauged by the effects it had on off-site members of the public and the environment and the standards of 10 C.F.R. § 70.61, was a low consequence event.

NFS clearly understands that the <u>on-site</u> consequences of a criticality accident could well be high and that every effort must be made to avoid such an accident. However, the potential <u>off-site</u> consequences of such an accident for members of the public and the environment, which are the areas of Intervenors greatest concern, are insignificant.

There are further inaccuracies in Intervenors' presentation that must be addressed. Intervenors state that exposures to members of the public as a result of the Tokai-Mura criticality accident would have been greater if the accident had not been brought under control. Intervenors' Pres. at 27. The statement is true but misleading. The Tokai-Mura criticality accident was not brought under control promptly and in fact emergency response at Tokai Mura was plagued by a complete lack of planning. The plant did not have a Criticality Accident Alarm System, (CAAS) and there was no formal emergency plan to deal with a criticality accident. As noted in the NRC Report, this led to "...a significant delay in development and communication of emergency protection measures for the public." (Page 3, Section 5.1) In fact, evacuation of residents within a 350 meter radius of the plant did not begin until 4.5 hours after the accident began. In contrast, NRC regulations in 10 CFR § 70.24 require licensees to have an adequate CAAS if they possess or handle more than 700 grams of ^{235}U. In 10 CFR 70.22, licensees who are required to have a CAAS are also required to have a detailed Emergency Plan. Thus, NFS has both a CAAS and an emergency plan.

The contrasts between the JCO facility and the NFS plant site in the areas of emergency preparedness and response are vivid. NFS has a CAAS that activates within 0.5 seconds of initiation of a criticality event. Routine drills demonstrate that all personnel evacuate to a safe assembly point within 5 minutes of CAAS activation. Trained radiation technicians begin immediate dose assessments, and assure that any areas receiving high doses are evacuated. The NFS guard force secures the facility, and walks down the site fenceline. The NFS Emergency Response Organization is functional within 15 minutes of CAAS activation. The lack of planning by JCO caused the long delay (20 hours) in bringing the system to a safe condition. By contrast, at NFS, the Emergency Response Organization is staffed with experts in all pertinent safety disciplines, including criticality safety. The organization would be aware of all pertinent site conditions and would be able to bring any accident situation under control much more rapidly. Controlling the accident quickly would provide a measure of protection for workers and the off-site public that was not present at Tokai-Mura.

In addition to NFS personnel controlling an accident, the Emergency Response Director would notify local agencies, including the Unicoi County Emergency Management Director, of the event and would provide emergency response recommendations (e.g., evacuation of nearby residents, instructions to stay indoors, etc). Local law enforcement agencies would secure nearby streets to stop incoming traffic. While evacuation of the area around the site would be a decision made by local authorities and might not be necessary if the accident were quickly brought under control, accident control and potential evacuation provide defense in depth to ensure that accident consequences to people are minimized. Again, such defense in depth was not present at Tokai-Mura.

Graded emergency response exercises are held at NFS every two years, with the involvement of all offsite agencies, to assure the emergency plan can be executed as intended in a timely manner. Thus, in addition to designing processes to assure a criticality event will not occur, NFS also has developed a comprehensive and effective emergency management plan to minimize the consequences in the highly unlikely event that an accident were to occur.

Intervenors also state that the consequences of the Tokai-Mura criticality accident would have been greater had the accident involved HEU (as opposed to intermediate enriched material). Intervenors' Pres. at 27. That is not necessarily the case. An accepted empirical model of the effects of criticality accidents, based on a review of data from historical accidents, relates the yield (number of fission events) to the volume of fissioning material (Olsen, A. R., R. L. Hooper, V. O. Uotinen, and C. L. Brown, "Empirical Model to Estimate Energy Released from Accidental Criticality," ANS Trans., 1974, **19**, 189-191). As the volume increases, yield increases; conversely, smaller volumes lead to smaller yields. Yield, i.e., the number of fission events, is directly related to offsite dose. The volume of fissile material required to achieve the critical state decreases with increasing enrichment. As a result, HEU can be made critical with a smaller volume of fissile material than can intermediate enriched uranium. Accordingly, at the point of criticality, an accident with HEU would involve a smaller volume of material than would an accident with intermediate enriched uranium. Therefore, in contrast to Intervenors claim, it is most probable that a criticality accident similar to the one at Tokai-Mura, but occurring with HEU, would lead to lower offsite doses.

Finally, Intervenors noted from the NRC Report that economic damages were estimated at $93 million. Intervenors' Pres. at 27. It is unclear how this is of any relevance. The $93 million sum was an estimate of what JCO expected to pay in compensation to nearby residents and businesses (Page 2, Section 4). The NRC Report does not discuss the purpose of these compensatory payments. Since there were no injuries, no physical damage to offsite structures, and no significant offsite contamination, it is hard to imagine their purpose. Nonetheless, the fact remains that the economic impact was borne entirely by the company, not the local community, as Intervenors imply.

In summary, a criticality accident in the BLEU facility is highly undesirable and would most likely have high consequences for any unfortunate onsite workers who were close by when the event occurred. Off-site consequences to members of the public and the environment, though, would almost certainly be low. This is evidenced by the dose data from the criticality accident at the JCO plant in Tokai-Mura. In the highly unlikely event a criticality accident were to occur at the BLEU facility, consequences to off-site members of the public and the environment would be <u>less</u> than those from the Tokai-Mura accident due to the existence of the detailed NFS emergency plan that includes provisions for bringing the accident under control as well as coordination with local emergency response organizations. Thus, the intervenors' discussion of the Tokai-Mura accident provides no basis for believing that a criticality accident at NFS would have significant consequences for either the off-site public or the environment.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 14, 2004.

Robert L. Frost

Robert L. Frost

John R. Frazier

I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 15, 2004.

Robert L. Frost

John R. Frazier

John R. Frazier