May 11, 1999

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PRIVATE FUEL STORAGE, L.L.C.	)	Docket No. 72-22-ISFSI
	)	
(Independent Spent Fuel Storage Installation)	)	
	)	

NRC STAFF'S RESPONSE TO APPLICANT'S MOTION FOR SUMMARY DISPOSITION OF UTAH CONTENTION C (DOSE LIMITS)

INTRODUCTION

Pursuant to 10 C.F.R. § 2.749(a), the NRC Staff ("Staff") herewith responds to the "Applicant's Motion for Summary Disposition of Utah Contention C" ("Motion"), filed on April 21, 1999 by Private Fuel Storage L.L.C. ("Applicant" or "PFS"). For the reasons set forth below and in the attached Affidavit of James Weldy and Elaine Keegan ("Weldy/Keegan Aff."), the Staff submits that each of the issues raised by Utah Contention C and its supporting basis statements have been resolved, and there no longer exists a genuine dispute of material fact with respect to this contention. Inasmuch as all issues have been resolved, the Applicant is entitled to a decision in its favor on Utah Contention C as a matter of law. The Staff therefore supports the Applicant's Motion and recommends that it be granted.

BACKGROUND

Utah Contention C ("Failure to Demonstrate Compliance with NRC Dose Limits") was filed by the State of Utah on November 23, 1997.[1] As admitted by the Licensing Board,[2] the contention states as follows:

> CONTENTION: The Applicant has failed to demonstrate a reasonable assurance that the dose limits specified in 10 CFR § 72.106(b) can and will be complied with in that:
>
> 1. License Application makes selective and inappropriate use of data from NUREG-1536 for the fission product release fraction.
>
> 2. License Application makes selective and inappropriate use of data from SAND80-2124 for the respirable particulate fraction.
>
> 3. The dose analysis in the License Application only considers dose due solely to inhalation of the passing cloud. Direct radiation and ingestion of food and water are not considered in the analysis.

In the basis statements for this contention, the State asserted that the Applicant's dose analysis in § 8.2.7.2 of its Safety Analysis Report ("SAR"), filed with its application of June 20, 1997, did not provide an adequate evaluation of the dose consequences of a loss-of-confinement accident at the Private Fuel Storage Facility ("PFSF"), in that it "makes selective and inappropriate use of data sources regarding doses, and fails to take important dose contributors into account" (Utah Contentions, at 18). Specifically, the State asserted (a) that the Applicant incorrectly assumed (in the table on SAR p. 8.2-37), that the fraction of Cs-134, Cs-137, and

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[1] "State of Utah's Contentions on the Construction and Operating License Application by Private Fuel Storage, LLC for an Independent Spent Fuel Storage Facility" ("Utah Contentions"), dated November 23, 1997, at 16-21.

[2] *Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), LBP-98-7 (1998), at 185-86.

Sr-90 that will be released into the canister is 2.3 E-5, based on NUREG-1536 ("Standard Review Plan for Dry Cask Storage Systems") (*Id.* at 19), (b) that the Applicant's dose analysis inappropriately relied upon a Sandia National Laboratories report concerning transportation accidents (SAND80-2124, "Transportation Accident Scenarios for Commercial Spent Fuel") to support its release fraction assumption that 90 % of the volatiles (Co-60, Sr-90, I-129, Ru-106, Cs-134 and Cs-137) released from the spent fuel to the canister will not escape the canister (*Id.*, *citing* SAR at 8.2-38); (c) that the Applicant's dose analysis inappropriately relied upon the Sandia report for its assumption that only 5% of the release fraction of Co-60 and Sr-90 will be respirable (*Id.* at 20, *citing* SAR at 8.2-39); and (d) that the Applicant's dose analysis failed to take into account the dose contributed by pathways other than inhalation of the passing cloud, such as direct radiation from cesium deposited on the ground, and ingestion of food and water or incidental soil ingestion, in violation of 10 C.F.R. § 72.24(m) (*Id.* at 21, *citing* SAR at 8.2-39).

In its motion for summary disposition of Utah Contention C, PFS asserts that the bases for the contention have been eliminated and that the contention is therefore no longer valid. In support of this assertion, PFS states that it has revised the challenged portions of its accident dose analysis in response to the Staff's RAIs and in accordance with ISG-5. In particular, PFS states that part 1 of the contention is no longer valid because its revised dose calculation no longer makes use of the fission product release fractions contained in NUREG-1536 or the assumptions in SAND80-2124 about the fraction of particulates or volatile fission products that would be released by the fuel but retained in the canister. Second, PFS states that part 2 of the contention is no longer valid because its revised dose calculation no longer makes use of the respirable particulate fraction contained in SAND80-2124. Third, PFS states that part 3 of the contention is no longer valid because its revised dose calculation takes into account all applicable environmental pathways to which a member of the public may be exposed both during passage of the contaminated plume and following deposition of contaminated material on the ground. *See* Motion for Summary Disposition, at 17-18; Affidavit of William Hennessy at 3-4. Accordingly, the Applicant concludes that summary disposition on Utah Contention C should be entered in its favor.

DISCUSSION

A. Legal Standards Governing Motions for Summary Disposition.

Pursuant to 10 C.F.R. § 2.749(a), "[a]ny party to a proceeding may move, with or without supporting affidavits, for a decision by the presiding officer in that party's favor as to all or any part of the matters involved in the proceeding. The moving party shall annex to the motion a separate, short, and concise statement of the material facts as to which the moving party contends that there is no genuine issue to be heard." In accordance with 10 C.F.R. § 2.749(b), when a properly supported motion for summary disposition is made, "a party opposing the motion may not rest upon the mere allegations or denials of his answer; his answer by affidavits or as otherwise provided in this section must set forth specific facts showing that there is a genuine issue of fact."[3] In addition, an opposing party must annex to its answer a short and concise statement

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[3] *Accord, Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units I and 2), ALAB-841, 24 NRC 64, 93 (1986). General denials and bare assertions are not sufficient to preclude summary disposition when the proponent of the motion has met its burden. *Advanced Medical Systems, Inc.* (One Factory Row, Geneva, Ohio 44041), CLI-93-22, 38 NRC 98, 102 (1993). Although the opposing party does not need to demonstrate that it will succeed on the issues, it must at least demonstrate that a genuine issue of fact exists to be tried. *Id.; Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-92-8, 35 NRC 145, 154 (1992) (to avoid summary disposition, the opposing party had to present contrary evidence that was so significantly probative as to create a material issue of fact).

of material facts as to which it contends there exists a genuine issue to be heard. 10 C.F.R. § 2.749(a). All material facts set forth in the moving party's statement will be deemed to be admitted unless controverted in the opposing party's statement. *Id.*

Pursuant to 10 C.F.R. § 2.49(d), "[t]he presiding officer shall render the decision sought if the filings in the proceeding, depositions, answers to interrogatories, and admissions on file, together with the statements of the parties and the affidavit, if any, show that there is no genuine issue as to any material fact and that the moving party is entitled to a decision as a matter of law."[4]

The Commission has encouraged the parties in its adjudicatory proceedings to utilize its summary disposition procedures "on issues where there is no genuine issue of material fact so that evidentiary hearing time is not unnecessarily devoted to such issues." Statement of Policy on Conduct of Licensing Proceedings, CLI-81-8, 13 NRC 452, 457 (1981).[5] Further, the Appeal Board has recognized that summary disposition provides "an efficacious means of avoiding unnecessary and possibly time-consuming hearings on demonstrably insubstantial issues."

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[4] Pursuant to 10 C.F.R. § 2.749(c), if a party opposing the motion demonstrates in its affidavits that valid reasons exist why it cannot provide facts essential to oppose the motion, the presiding officer may deny the motion, order a continuance to permit affidavits to be obtained, or take such other action as may be appropriate.

[5] The Commission recently endorsed its earlier policy statement, but indicated that "Boards should forego the use of motions for summary disposition except upon a written finding that such a motion will likely substantially reduce the number of issues to be decided, or otherwise expedite the proceeding." *Statement of Policy on Conduct of Adjudicatory Proceedings*, CLI-98-12, 48 NRC 18, 20-21 (1998). The Commission has also expressed its satisfaction with the Licensing Board's expeditious handling of this proceeding. *Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-98-13, 48 NRC 26, 37 (1998). The Staff submits that summary disposition of Utah Contention C will reduce the multiplicity of issues that require hearings in this proceeding, and will otherwise serve to expedite the proceeding.

Wisconsin Electric Power Co. (Point Beach Nuclear Plant, Unit 1), ALAB-696, 16 NRC 1245, 1263 (1982); *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 NRC 542, 550 (1980).[6]

The Commission's summary disposition procedures have been analogized to Rule 56 of the Federal Rules of Civil Procedure. *See, e.g., Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-443, 6 NRC 741, 753-54 (1977). Indeed, the Commission, when considering motions for summary disposition filed pursuant to 10 C.F.R. § 2.749, generally applies the same standards that the Federal courts use in determining motions for summary judgment under Rule 56 of the Federal Rules. *Advanced Medical Systems,* 38 NRC at 102 (1993). Decisions arising under Rule 56 of the Federal Rules may thus serve as guidelines to the Commission's adjudicatory boards in applying 10 C.F.R. § 2.749. *Perry, supra,* 6 NRC at 754.

Under Rule 56 of the Federal Rules, the party seeking summary judgment has the burden of proving the absence of genuine issues of material fact. *Adickes v. S. H. Kress & Co.*, 398 U.S. 144, 157 (1970); *Advanced Medical Systems,* 38 NRC at 102. In addition, the record is viewed in the light most favorable to the party opposing the motion. *Poller v. CBS, Inc.*, 368 U.S. 464, 473 (1962); *Kerr-McGee Chemical Corp.* (West Chicago Rare Earths Facility), ALAB-944, 33 NRC 81, 144 (1991). However, if the moving party makes a proper showing for summary disposition and the opposing party fails to show that there is a genuine issue of material fact, the

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[6] It is well settled that an agency may ordinarily dispense with an evidentiary hearing where no genuine issue of material fact exists. *Veg-Mix, Inc.* v. *U.S. Dep't of Agriculture*, 832 F.2d 601, 607-08 (D.C. Cir. 1987).

District Court (or Licensing Board) may summarily dispose of all of the matters before it on the basis of the filings in the proceeding, the statements of the parties, and affidavits. Rule 56 (e), Fed. R. Civ. P. *Accord, Advanced Medical Systems,* 38 NRC at 102; 10 C.F.R. § 2.749(d).

For the reasons set forth below and in the attached affidavit, the Staff submits that in the instant proceeding, there does not exist any genuine issue of material fact with respect to Utah Contention C, and the Applicant is entitled to a decision in its favor on this contention as a matter of law.

B. <u>Adequacy of the Applicant's Loss-of-Confinement Dose Analysis.</u>

1. <u>Applicable Regulatory Standards.</u>

Pursuant to 10 C.F.R. § 72.106(b) (as revised in October 1998, to be consistent with 10 C.F.R. Part 20 dose calculational methodology (63 Fed. Reg. 54559)), an applicant for an independent spent fuel storage installation (ISFSI) must establish a controlled area such that:

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent shall not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or to any extremity shall not exceed 0.5 Sv (50 rem). . . .

Also, as set forth in 10 C.F.R. § 72.24(m), an applicant's SAR is required to contain:

> An analysis of the potential dose equivalent or committed dose equivalent to an individual outside the controlled area from accidents or natural phenomena events that result in the release of radioactive material to the environment or direct radiation from the ISFSI . . . . The calculations of individual dose equivalent or committed dose equivalent must be performed for direct exposure, inhalation, and ingestion occurring as a result of the postulated design basis event.

Further, as set forth in 10 C.F.R. § 72.126(d), an applicant is required, *inter alia,* to submit analyses of design basis accidents which "show that releases to the general environment will be within the exposure limits given in § 72.106."

The NRC Staff has issued various guidance documents concerning the proper methodology for calculating offsite doses for design basis events. Certain guidance is contained, for example, in NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage, at 15-32 (Draft, September 1998). More recently (and subsequent to the Applicant's submittal of its SAR), the Staff issued further guidance on the proper methodology to be utilized in calculating offsite doses resulting from a loss-of-confinement accident, as set forth in Interim Staff Guidance-5 (ISG-5), entitled "Accident Dose Calculations" (September 28, 1998). ISG-5 recommends the use of release fractions contained in NUREG-1617, Standard Review Plan for Transportation Packages for Spent Nuclear Fuel (DRAFT, March 1998), Table 4-1. In addition, ISG-5 describes an acceptable method to account for radionuclides that are released into the cask volume but do not escape the cask volume based on the leakage rate of air out of a small hole in the confinement boundary. The technical bases for these release fractions (pertaining to the release of gases, volatiles and particulate from the fuel to the cask interior) and calculation methodology are described in NUREG/CR-6487, Containment Analysis for Type B Packages to Transport Various Contents (November 1996). In contrast to previous Staff guidance, ISG-5 does not assume that the confinement boundary will be breached (non-mechanistic failure). This is consistent with structural analysis which demonstrates that the confinement integrity is maintained during normal, off-normal, and accident conditions. Also, ISG-5 recommends the use of larger values for the concentration of "CRUD" on BWR fuel, and consideration of a more comprehensive array of gaseous, volatile, and particulate radionuclides in the calculation. ISG-5 does not include any mitigation of the radioactive source term available for release from the cask interior to the environment -- *i.e.,* no credit is given for plateout, particle size, etc. This provides a bounding condition for the analysis. *See* Weldy/Keegan Aff. at 5-6.

2. The Applicant's Revised Dose Analysis.

As set forth in the attached affidavit, the Staff has reviewed the revised accident dose calculation which PFS submitted to the NRC in its February 1999 response to the Staff's RAIs. On the basis of this review, the Staff has determined that the Applicant's revised dose analysis satisfactorily addresses each of the concerns raised by this contention, that it appropriately follows the guidance in ISG-5, and that its resulting dose estimates satisfy the regulatory requirements set forth in 10 C.F.R. Part 72. Accordingly, the Staff has concluded that upon revision of the SAR to reflect the Applicant's revised dose analysis, the license application will satisfy the Commission's regulatory requirements pertaining to the analysis of offsite dose consequences of a loss-of-confinement accident. Weldy/Keegan Aff. at 4.[7]

In its initial SAR, the Applicant utilized the release fractions from NUREG-1536, Table 7.1, to estimate the quantity of radioactive material that is released from the fuel into the cask cavity during a loss-of-confinement accident. The Applicant then reduced this release

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[7] The Applicant has indicated that it intends to revise its SAR to incorporate its revised dose analysis. *See* "Applicant's Response to State of Utah's Proprietary and Non-Proprietary Motions to Compel Applicant to Respond to State's First Set of Discovery Requests," dated May 7, 1999, at 6 n.12 ("PFS intends to file a license amendment on or about May 14, 1999 which will formally incorporate into the License Application the various analyses and commitments that it has made in its RAI responses filed in February . . ."). The Staff understands that this submittal may be delayed for several days, to on or about May 19, 1999.

quantity by (a) the fraction of volatile and particulate material that plates out on the interior of the cask and is not released to the environment, and (b) the fraction of Sr-90 and Co-60 that are not respirable and cannot contribute to the inhalation dose, which the applicant obtained from SAND80-2124. *Id.* at 6-7.

In section 8.2.7 of its initial SAR, the Applicant included a calculation of the consequences from a postulated loss-of-confinement accident, which considered the committed effective dose equivalent (CEDE) from the inhalation of the passing cloud; the SAR did not calculate the doses received by members of the public from other pathways, such as from direct exposure and ingestion, as required by 10 C.F.R. § 72.24(m). *Id.* at 7.

The Applicant's accident dose analysis was the subject of two separate Requests for Additional Information (RAIs) transmitted by the Staff to PFS. On April 1, 1998, the Staff requested additional information concerning the Applicant's accident dose calculations, including the basis for its assumption of a respirable fraction of 5 percent for Co-60 and Sr-90, and its consideration of an inhalation pathway only (*see* RAIs 8-4, 8-5, and 8-8, dated April 1, 1998). Subsequently, the Applicant's dose analysis was further addressed in an RAI transmitted by the Staff to PFS on December 10, 1998. In particular, RAIs 7-1 and 8-4 of this second round of RAIs requested that the Applicant revise its dose calculations to correct its assumptions for the respirable fraction of Co-60 released in an accident; that the Applicant follow the latest NRC guidance on calculating offsite doses for a loss-of-confinement accident, as set forth in Interim Staff Guidance 5 (ISG-5), "Accident Dose Calculations"; and that the Applicant justify its failure to model pathways other than the inhalation pathway. Responses to the Staff's first and second round RAIs concerning these matters were submitted by PFS on May 19, 1998, and February 10, 1999, respectively. PFS submitted a partial revision to Chapter 8 of its SAR on May 22, 1998; and has indicated that it will submit a further SAR revision in May 1999 that will incorporate its revised accident dose analysis, discussed below. *Id.* at 7-8; *see* n.7, *supra*.

The Applicant's revised dose analysis, set forth in its February 10, 1999 response to RAIs and its February 1999 revision of its SAR, appropriately take into account the considerations set forth in ISG-5, with respect to the respirable release fractions of radionuclides and mitigation factors such as plateout and deposition. The Applicant's revised dose analysis conservatively assumes that 100% of the released radioactive material is respirable. The revised dose analysis bases the release quantity of radioactive material from the free volume inside the cask to the exterior of the cask on the volume of air that can leak through a very small diameter hole assumed to exist in the containment boundary under accident conditions, consistent with ISG-5. This methodology is in contrast to the Applicant's original accident dose calculation in that it does not rely on a constant fraction of mass released from the fuel that escapes containment to account for mitigation factors such as plateout and deposition of material within the breached cask. *Id.* at 8.

In light of the Applicant's revised dose analysis, Part 1 of Utah Contention C, which asserted that the license application made selective and inappropriate use of data from NUREG-1536 for the fission product release fraction, is no longer applicable, because (1) the accident dose calculation no longer utilizes data from NUREG-1536 for the fission product release fraction; (2) the accident dose calculation no longer utilizes data from SAND80-2124 for the fission product release fraction; and (3) the accident dose calculation follows a single NRC guidance document (ISG-5) and does not make selective and inappropriate use of data from any source. Accordingly, Part 1 of Utah Contention C is no longer valid. *See id.* at 8-9.

Similarly, the Applicant's revised dose calculation no longer takes credit for any reduction in dose due to the size distribution of the released particulates. The original dose calculation utilized data from SAND80-2124 to support its assumption that only five percent of the isotopes Co-60 and Sr-90 released from the fuel assemblies will be respirable by a human. The revised dose calculation assumes that all particulates matter released from the cask will be respirable. Therefore, Part 2 of Utah Contention C, which asserted that the License Application made selective and inappropriate use of data from SAND80-2124 for the respirable particulate fraction, is no longer applicable because (1) the accident dose calculation no longer utilizes data from SAND80-2124 for the respirable particulate fraction; and (2) the accident dose calculation follows a single NRC guidance document (ISG-5) and therefore does not make selective and inappropriate use of data from any source. Accordingly, Part 2 of Utah Contention C is no longer valid. *See id.* at 9.

The Applicant's revised dose analysis also addresses the concerns raised in part 3 of the contention, with respect to dose pathways. In its revised dose analysis, the applicant has included an assessment of the dose delivered to members of the public following the deposition on the ground of radioactive material in the plume from a loss-of-confinement accident. This is in accordance with the requirements of 10 CFR 72.24(m), which requires that calculations of individual dose equivalent or committed dose equivalent be performed for direct exposure, inhalation, and ingestion occurring as a result of postulated design basis events. The revised dose calculation assesses the dose received by a receptor from the direct exposure to contaminated ground, inhalation of resuspended radioactive material, ingestion of milk and beef following grazing of contaminated plants, and inadvertent ingestion of soil contaminated with radioactive material deposited on the ground. Additionally, the revised dose calculation determines the dose received from the external exposure to the contaminated plume as it passes the receptor. While the revised analysis omits the surface water and groundwater pathways, this is not inappropriate, based on the Applicant's determination, described in its Environmental Report, that there are no public or private surface drinking water supplies in the PFSF vicinity and there are no wells used for drinking water located near the boundary of the controlled area of the ISFSI, which is the location at which a member of the public could receive the greatest dose from the accident. *Id.* at 9-10.

The Applicant's revised dose analysis includes dose calculations for a receptor located at the PFSF site boundary and at a location representing the nearest actual residences to the facility using realistic estimates of exposure times for receptors located at both locations. Both locations showed that the dose following deposition of radioactive material in the soil was dominated by external exposure to Co-60. The Applicant's calculations also showed that a higher dose was received by an individual located at the PFSF fence than by individuals located at actual residences in the area. *Id.* at 10.

Based on the Staff's review of the Applicant's revised dose analysis, as set forth in its February 1999 response to the Staff's RAIs, the Staff has determined that the pathways considered by the Applicant are appropriate and adequate to assess the dose that an individual located at the site boundary would receive from the passing cloud and following the deposition of radioactive material on the ground after a loss-of-confinement accident. Further, the Staff agrees with the Applicant's determination that an individual located at the site boundary would be the member of the public who would receive the largest dose from a loss-of-confinement accident. *Id.* at 11.

In light of the Applicant's revised dose analysis, part 3 of Utah Contention C is no longer valid, in that (1) the revised dose calculation determines the dose from direct exposure to the maximally exposed member of the public from the contaminated plume of airborne radioactive material; (2) the revised dose calculation determines the dose from all applicable pathways for the maximally exposed member of the public including direct exposure, inhalation, and ingestion pathways from the soil contaminated by radioactive material deposited by the plume; and (3) the ingestion of contaminated water is not a credible pathway because the member of the public who would receive the largest dose from a loss-of-confinement accident is a hypothetical individual located just outside the site boundary -- and there are no permanent residences, or public or private surface drinking water supplies or wells used for drinking water at the location of the maximally exposed individual. *Id.*

Based on the Staff's review of the Applicant's revised dose analysis, the Staff is satisfied that the revised dose calculation was performed in accordance with applicable Staff guidance, contained in ISG-5, and that it satisfies applicable NRC requirements. Specifically, the revised dose analysis meets the requirements of 10 CFR 72.24(m), by performing calculations of individual dose equivalent for direct exposure, inhalation, and ingestion occurring as a result of a loss-of-confinement accident. Further, the revised dose calculation meets the requirements of 10 CFR 72.106(b), by demonstrating that any individual located on or beyond the nearest boundary of the controlled area will not receive from a loss-of-confinement accident a total effective dose equivalent of 0.05 Sv (5 rem). Additionally, if the entire dose calculated by the Applicant was deposited in any single organ, the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye would not exceed 0.5 Sv (50 rem); the lens dose equivalent would not exceed 0.15 Sv (15 rem); and the shallow dose equivalent to skin or any extremity would not exceed 0.5 Sv (50 rem). *Id.* at 11-12.

Based upon the above considerations, the Staff has concluded that upon revision of the SAR to reflect the Applicant's revised dose analysis, which the Applicant indicates will be submitted later this month, the license application will satisfy the Commission's regulatory requirements pertaining to the analysis of offsite dose consequences of a loss-of-confinement accident. Further, upon revision of the SAR to reflect the Applicant's revised dose analysis, there is no basis for Utah Contention C. *Id.* at 12.

CONCLUSION

Based upon the above considerations, as set forth in the attached Affidavit, the Staff has concluded that upon revision of the SAR to reflect the Applicant's revised dose analysis, the license application will satisfy the Commission's regulatory requirements pertaining to the analysis of offsite dose consequences of a loss-of-confinement accident. Further, upon the Applicant's revision of its SAR to reflect its revised dose analysis, there is no longer any basis for Utah Contention C. Accordingly, the Staff submits that a decision in the Applicant's favor on Utah Contention C is warranted as a matter of law.

Respectfully submitted,

Sherwin E. Turk

Sherwin E. Turk
Counsel for NRC Staff

Dated at Rockville, Maryland
this 11th day of May, 1999

May 11, 1999

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
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PRIVATE FUEL STORAGE, L.L.C.	)	Docket No. 72-22-ISFSI
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(Independent Spent Fuel Storage Installation)	)	
	)	

AFFIDAVIT OF JAMES WELDY AND ELAINE KEEGAN
CONCERNING UTAH CONTENTION C (DOSE LIMITS)

James Weldy (JW) and Elaine Keegan (EK), having first been duly sworn, do hereby state as follows:

1(a). (EK) My name is Elaine Keegan. I am employed as a Health Physicist in the Technical Review Directorate, Spent Fuel Project Office, Office of Nuclear Materials Safety and Safeguards, U.S. Nuclear Regulatory Commission (NRC), in Washington, D.C. A statement of my professional qualifications is attached hereto.

1(b). (JW) My name is James Weldy. I am employed as a Research Engineer at the Center for Nuclear Waste Regulatory Analyses (CNWRA), which is a division of the Southwest Research Institute (SWRI), in San Antonio, Texas. I am providing this affidavit under a technical assistance contract between the NRC Staff and the SWRI. A statement of my professional qualifications is attached hereto.

2. This Affidavit is prepared in response to the "Applicant's Motion for Summary Disposition of Utah Contention C," filed on April 21, 1999 by Private Fuel

Storage L.L.C. ("Applicant" or "PFS"), along with the Affidavit of William Hennessy, dated April 21, 1999.

3. Utah Contention C, entitled "Failure to Demonstrate Compliance with NRC Dose Limits," states as follows:

> CONTENTION: The Applicant has failed to demonstrate a reasonable assurance that the dose limits specified in 10 CFR § 72.106(b) can and will be complied with in that:
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> 1. License Application makes selective and inappropriate use of data from NUREG-1536 for the fission product release fraction.
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> 2. License Application makes selective and inappropriate use of data from SAND80-2124 for the respirable particulate fraction.
>
> 3. The dose analysis in the License Application only considers dose due solely to inhalation of the passing cloud. Direct radiation and ingestion of food and water are not considered in the analysis.

4. Utah Contention C refers to the Applicant's calculation of the dose that would be received by a member of the public in the event of a loss-of-confinement accident at the Private Fuel Storage Facility ("PFSF"), as presented in the Applicant's Safety Analysis Report ("SAR") that was filed with its application of June 20, 1997.

5. In the basis statements for Utah Contention C, the State of Utah asserted that the Applicant's dose analysis in § 8.2.7.2 of the SAR did not provide an adequate evaluation of the dose consequences of a loss-of-confinement accident in that it "makes selective and inappropriate use of data sources regarding doses, and fails to take important dose contributors into account" (Utah Contentions, at 18). Specifically, the State asserted

(a) that the Applicant incorrectly assumed (in the table on SAR p. 8.2-37), that the fraction of Cs-134, Cs-137, and Sr-90 that will be released into the canister is 2.3 E-5, based on NUREG-1536, Standard Review Plan for Dry Cask Storage Systems (January 1997) (*Id.* at 19), (b) that the Applicant's dose analysis inappropriately relied upon a Sandia National Laboratories report concerning transportation accidents (SAND80-2124, Transportation Accident Scenarios for Commercial Spent Fuel (1981)), to support its release fraction assumption that 90 % of the volatiles (Co-60, Sr-90, I-129, Ru-106, Cs-134 and Cs-137) released from the spent fuel to the canister will not escape the canister (*Id., citing* SAR at 8.2-38); (c) that the Applicant's dose analysis inappropriately relied upon the Sandia report for its assumption that only 5% of the release fraction of Co-60 and Sr-90 will be respirable (*Id.* at 20, *citing* SAR at 8.2-39); and (d) that the Applicant's dose analysis failed to take into account the dose contributed by pathways other than inhalation of the passing cloud, such as direct radiation from cesium deposited on the ground, and ingestion of food and water or incidental soil ingestion, in violation of 10 CFR § 72.24(m) (*Id.* at 21, *citing* SAR at 8.2-39).

6. In its motion for summary disposition of Utah Contention C, PFS asserts that the bases for the contention have been eliminated and that the contention is therefore no longer valid. In support of this assertion, PFS states that it has revised the challenged portions of its accident dose analysis, in response to the Staff's RAIs, and that its revised calculation was performed in accordance with ISG-5. In particular, PFS states that part 1 of the contention is no longer valid because its revised dose calculation no longer makes use of the fission product release fractions contained in NUREG-1536 or the assumptions

in SAND80-2124 about the fraction of particulates or volatile fission products that would be released by the fuel but retained in the canister. Second, PFS states that part 2 of the contention is no longer valid because its revised dose calculation no longer makes use of the respirable particulate fraction contained in SAND80-2124. Third, PFS states that part 3 of the contention is no longer valid because its revised dose calculation takes into account all applicable environmental pathways to which a member of the public may be exposed both during passage of the contaminated plume and following deposition of contaminated material on the ground. *See* Applicant's Motion at 17-18; Affidavit of William Hennessy at 3-4.

7. We have reviewed the Applicant's revised accident dose calculation, which PFS submitted to the NRC in its February 1999 response to the Staff's RAIs. On the basis of our review, we believe that the Applicant's revised dose analysis satisfactorily addresses each of the concerns raised by this contention. Further, we are satisfied that the Applicant's revised accident dose analysis appropriately follows the guidance in ISG-5, and that its resulting dose estimates satisfy the regulatory requirements set forth in 10 C.F.R. Part 72. Accordingly, we are satisfied that upon revision of the SAR to reflect the Applicant's revised dose analysis, the license application will satisfy the Commission's regulatory requirements pertaining to the analysis of offsite dose consequences of a loss-of-confinement accident. The bases for these conclusions are as follows.

8. Pursuant to 10 C.F.R. § 72.106(b) (as revised in October 1998, to be consistent with 10 C.F.R. Part 20 dose calculational methodology (63 Fed. Reg. 54559)),

an applicant for an independent spent fuel storage installation (ISFSI) must establish a controlled area such that:

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent shall not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or to any extremity shall not exceed 0.5 Sv (50 rem). . . .

Also, as set forth in 10 C.F.R. § 72.24(m), an applicant's SAR is required to contain:

> An analysis of the potential dose equivalent or committed dose equivalent to an individual outside the controlled area from accidents or natural phenomena events that result in the release of radioactive material to the environment or direct radiation from the ISFSI . . . . The calculations of individual dose equivalent or committed dose equivalent must be performed for direct exposure, inhalation, and ingestion occurring as a result of the postulated design basis event.

Further, as set forth in 10 C.F.R. § 72.126(d), an applicant is required, *inter alia,* to submit analyses of design basis accidents which "show that releases to the general environment will be within the exposure limits given in § 72.106."

9. The NRC Staff has issued various guidance documents concerning the proper methodology for calculating offsite doses for design basis events. Certain guidance is contained, for example, in NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage, at 15-32 (Draft, September 1998). More recently (and subsequent to the Applicant's submittal of its SAR), the Staff issued further guidance on the proper

methodology to be utilized in calculating offsite doses resulting from a loss-of-confinement accident, as set forth in Interim Staff Guidance-5 (ISG-5), entitled "Accident Dose Calculations" (September 28, 1998). ISG-5 recommends the use of release fractions contained in NUREG-1617, Standard Review Plan for Transportation Packages for Spent Nuclear Fuel (DRAFT, March 1998), Table 4-1. In addition, ISG-5 describes an acceptable method to account for radionuclides that are released into the cask volume but do not escape the cask volume based on the leakage rate of air out of a small hole in the confinement boundary. The technical bases for these release fractions (pertaining to the release of gases, volatiles and particulate from the fuel to the cask interior) and calculation methodology are described in NUREG/CR-6487, Containment Analysis for Type B Packages to Transport Various Contents (November 1996). In contrast to previous Staff guidance, ISG-5 does not assume that the confinement boundary will be breached (non-mechanistic failure). This is consistent with structural analysis which demonstrates that the confinement integrity is maintained during normal, off-normal, and accident conditions. Also, ISG-5 recommends the use of larger values for the concentration of "CRUD" on BWR fuel, and consideration of a more comprehensive array of gaseous, volatile, and particulate radionuclides in the calculation. ISG-5 does not include any mitigation of the radioactive source term available for release from the cask interior to the environment -- *i.e.*, no credit is given for plateout, particle size, etc. This provides a bounding condition for the analysis.

10. In its initial SAR, the Applicant utilized the release fractions from NUREG-1536, Table 7.1, to estimate the quantity of radioactive material that is released

from the fuel into the cask cavity during a loss-of-confinement accident. The Applicant then reduced this release quantity by (a) the fraction of volatile and particulate material that plates out on the interior of the cask and is not released to the environment, and (b) the fraction of Sr-90 and Co-60 that are not respirable and cannot contribute to the inhalation dose, which the applicant obtained from SAND80-2124.

11. In section 8.2.7 of its initial SAR, the Applicant included a calculation of the consequences from a postulated loss-of-confinement accident, which considered the committed effective dose equivalent (CEDE) from the inhalation of the passing cloud; the SAR did not calculate the doses received by members of the public from other pathways, such as from direct exposure and ingestion, as required by 10 C.F.R. § 72.24(m).

12. The Applicant's accident dose analysis was the subject of two separate Requests for Additional Information (RAIs) transmitted by the Staff to PFS. On April 1, 1998, the Staff requested additional information concerning the Applicant's accident dose calculations, including the basis for its assumption of a respirable fraction of 5 percent for Co-60 and Sr-90, and its consideration of an inhalation pathway only (*see* RAIs 8-4, 8-5, and 8-8, dated April 1, 1998). Subsequently, the Applicant's dose analysis was further addressed in an RAI transmitted by the Staff to PFS on December 10, 1998. In particular, RAIs 7-1 and 8-4 of this second round of RAIs requested that the Applicant revise its dose calculations to correct its assumptions for the respirable fraction of Co-60 released in an accident; that the Applicant follow the latest NRC guidance on calculating offsite doses for a loss-of-confinement accident, as set forth in ISG-5; and that the Applicant justify its failure to model pathways other than the inhalation pathway. Responses to the Staff's first

and second round RAIs concerning these matters were submitted by PFS on May 19, 1998, and February 10, 1999, respectively. PFS submitted a partial revision to Chapter 8 of its SAR on May 22, 1998; and we understand that PFS has indicated it will submit a further SAR revision in May 1999 that will incorporate its revised accident dose analysis, discussed below.

13. The Applicant's revised dose analysis, set forth in its February 10, 1999 response to RAIs and its February 1999 revision of its SAR, appropriately takes into account the considerations set forth in ISG-5, with respect to the respirable release fractions of radionuclides and mitigation factors such as plateout and deposition. The Applicant's revised dose analysis conservatively assumes that 100% of the released radioactive material is respirable. The revised dose analysis bases the release quantity of radioactive material from the free volume inside the cask to the exterior of the cask on the volume of air that can leak through a very small diameter hole assumed to exist in the containment boundary under accident conditions, consistent with ISG-5. This methodology is in contrast to the Applicant's original accident dose calculation in that it does not rely on a constant fraction of mass released from the fuel that escapes containment to account for mitigation factors such as plateout and deposition of material within the breached cask.

14. In light of the Applicant's revised dose analysis, Part 1 of Utah Contention C, which asserted that the license application made selective and inappropriate use of data from NUREG-1536 for the fission product release fraction, is no longer applicable, because (1) the accident dose calculation no longer utilizes data from NUREG-1536 for the fission product release fraction; (2) the accident dose calculation no

longer utilizes data from SAND80-2124 for the fission product release fraction; and (3) the accident dose calculation follows a single NRC guidance document (ISG-5) and does not make selective and inappropriate use of data from any source. Accordingly, Part 1 of Utah Contention C is no longer valid.

15. Similarly, the Applicant's revised dose calculation no longer takes credit for any reduction in dose due to the size distribution of the released particulates. The original dose calculation utilized data from SAND80-2124 to support its assumption that only five percent of the isotopes Co-60 and Sr-90 released from the fuel assemblies will be respirable by a human. The revised dose calculation assumes that all particulates matter released from the cask will be respirable. Therefore, Part 2 of Utah Contention C, which asserted that the License Application made selective and inappropriate use of data from SAND80-2124 for the respirable particulate fraction, is no longer applicable because (1) the accident dose calculation no longer utilizes data from SAND80-2124 for the respirable particulate fraction; and (2) the accident dose calculation follows a single NRC guidance document (ISG-5) and therefore does not make selective and inappropriate use of data from any source. Accordingly, Part 2 of Utah Contention C is no longer valid.

16. The Applicant's revised dose analysis also addresses the concerns raised in part 3 of the contention, with respect to dose pathways. In its revised dose analysis, the applicant has included an assessment of the dose delivered to members of the public following the deposition on the ground of radioactive material in the plume from a loss-of-confinement accident. This is in accordance with the requirements of 10 CFR 72.24(m), which requires that calculations of individual dose equivalent or committed dose equivalent

be performed for direct exposure, inhalation, and ingestion occurring as a result of postulated design basis events. The revised dose calculation assesses the dose received by a receptor from the direct exposure to contaminated ground, inhalation of resuspended radioactive material, ingestion of milk and beef following grazing of contaminated plants, and inadvertent ingestion of soil contaminated with radioactive material deposited on the ground. Additionally, the revised dose calculation determines the dose received from the external exposure to the contaminated plume as it passes the receptor. While the revised analysis omits the surface water and groundwater pathways, this is not inappropriate, based on the Applicant's determination, described in its Environmental Report, that there are no public or private surface drinking water supplies in the PFSF vicinity and there are no wells used for drinking water located near the boundary of the controlled area of the ISFSI, which is the location at which a member of the public could receive the greatest dose from the accident.

17. The Applicant's revised dose analysis includes dose calculations for a receptor located at the PFSF site boundary and at a location representing the nearest actual residences to the facility using realistic estimates of exposure times for receptors located at both locations. Both locations showed that the dose following deposition of radioactive material in the soil was dominated by external exposure to Co-60. The Applicant's calculations also showed that a higher dose was received by an individual located at the PFSF fence than by individuals located at actual residences in the area.

18. Based on our review of the Applicant's revised dose analysis, as set forth in its February 1999 response to the Staff's RAIs, we believe the pathways considered by

the Applicant are appropriate and adequate to assess the dose that an individual located at the site boundary would receive from the passing cloud and following the deposition of radioactive material on the ground after a loss-of-confinement accident. Further, we agree with the Applicant's determination that an individual located at the site boundary would be the member of the public who would receive the largest dose from a loss-of-confinement accident.

19. In light of the Applicant's revised dose analysis, part 3 of Utah Contention C is no longer valid, in that (1) the revised dose calculation determines the dose from direct exposure to the maximally exposed member of the public from the contaminated plume of airborne radioactive material; (2) the revised dose calculation determines the dose from all applicable pathways for the maximally exposed member of the public including direct exposure, inhalation, and ingestion pathways from the soil contaminated by radioactive material deposited by the plume; and (3) the ingestion of contaminated water is not a credible pathway because the member of the public who would receive the largest dose from a loss-of-confinement accident is a hypothetical individual located just outside the site boundary -- and there are no permanent residences, or public or private surface drinking water supplies or wells used for drinking water at the location of the maximally exposed individual.

20. Based on our review of the Applicant's revised dose analysis, we are satisfied that the revised dose calculation was performed in accordance with applicable Staff guidance, contained in ISG-5, and that it satisfies applicable NRC requirements. Specifically, the revised dose analysis meets the requirements of 10 CFR 72.24(m), by

performing calculations of individual dose equivalent for direct exposure, inhalation, and ingestion occurring as a result of a loss-of-confinement accident. Further, the revised dose calculation meets the requirements of 10 CFR 72.106(b), by demonstrating that any individual located on or beyond the nearest boundary of the controlled area will not receive from a loss-of-confinement accident a total effective dose equivalent of 0.05 Sv (5 rem). Additionally, if the entire dose calculated by the Applicant was deposited in any single organ, the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye would not exceed 0.5 Sv (50 rem); the lens dose equivalent would not exceed 0.15 Sv (15 rem); and the shallow dose equivalent to skin or any extremity would not exceed 0.5 Sv (50 rem).

21. Based upon the above considerations, we have concluded that upon revision of the SAR to reflect the Applicant's revised dose analysis, the license application will satisfy the Commission's regulatory requirements pertaining to the analysis of offsite dose consequences of a loss-of-confinement accident. Further, upon revision of the SAR to reflect the Applicant's revised dose analysis, there is no basis for Utah Contention C.

22. I hereby certify that the foregoing is true and correct to the best of my knowledge, information and belief.

Elaine Keegan

Elaine Keegan

Subscribed and sworn to before me
this 11th day of May, 1999.

CIRCE E. MARTIN
NOTARY PUBLIC
MONTGOMERY CO., MD.

Circe E. Martin

Notary Public

My commission expires: March 1, 2003

James Weldy

James Weldy

CIRCE E. MARTIN
NOTARY PUBLIC
MONTGOMERY CO., MD.

Subscribed and sworn to before me
this 11th day of May, 1999.

Circe E. Martin

Notary Public

My commission expires: March 1, 2003

James Weldy

Research Engineer

Center for Nuclear Waste Regulatory Analyses

B.S. in Nuclear Engineering, The University of Michigan, 1995
M.Eng. in Radiological Health Engineering, The University of Michigan, 1996

Mr. Weldy has experience in many areas of Radiological Health and Nuclear Engineering. He is skilled in environmental transport and pathway analysis, radiation dose calculation, risk assessment, radiation shielding analysis, criticality assessment, and detection and measurement of radiation.

During his practicum for his Master's degree at Argonne National Laboratory, he designed a Bonner sphere system to measure the dose delivered to a person by a pulsed neutron source. This included writing a Monte Carlo code to model the transport of neutrons through the spheres and the temporal arrival of the neutrons at the detector.

He is currently performing calculations and modeling to assess the performance of a proposed high-level nuclear waste repository at Yucca Mountain, Nevada. His responsibilities at the CNWRA include conducting research into dose assessments for the proposed high-level radioactive waste disposal site. He also provides technical support to the Nuclear Regulatory Commission for projects including uranium recovery activities related to *in situ* leach mining, evaluating reclamation plans and license amendment requests, and assessing interim storage of spent nuclear fuel including evaluating safety analysis reports and environmental reports. He has been responsible for the evaluation of radiation hazards to workers and members of the public from Independent Spent Fuel Storage Installations and has been involved in the licensing of these facilities. He has assisted in the preparation of environmental impact statements and evaluations of the hazards associated with the wastes contained in the high-level waste tanks at the Hanford Site in Washington state.

PROFESSIONAL CHRONOLOGY: Research Fellow, Argonne National Laboratory, 1996; Engineer, Southwest Research Institute, Center for Nuclear Waste Regulatory Analyses, 1997-1998; Research Engineer, Southwest Research Institute, Center for Nuclear Waste Regulatory Analyses, 1998-present

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B.S. in Radiological Health Physics, University of Lowell, 1978

Ms. Keegan has experience in many areas of Radiological Health Physics. She is skilled in radiation shielding analysis, environmental transport and pathway analysis, radiation dose calculation, and emergency planning.

Ms. Keegan is currently performing shielding evaluations for the licensing of spent nuclear fuel transportation and storage casks. Her work includes the evaluation of accident dose analyses, occupational dose calculations, and the adequacy of proposed radiation protection programs. She has provided input for the preparation of NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities; NUREG-1536, Standard Review Plan for Dry Cask Storage Systems; and NUREG-1617, Standard Review Plan for Transportation Packages for Spent Nuclear Fuel. She has reviewed emergency plans for dry storage facilities. She has also prepared several environmental assessments for licensing actions involving the storage of spent fuel.

Ms. Keegan also has been project manager for the license renewal of a major fuel fabrication facility, which included evaluating criticality, radiation protection, emergency planning, fire safety, and environmental protection. She has prepared numerous environmental assessments for the renewal of fuel cycle facilities, and she has assisted in the preparation of several environmental impact statements relating to the fuel fabrication facilities.

While working at the Vermont Yankee nuclear power plant, Ms. Keegan was responsible for implementing the facility's radiological environmental monitoring program. She prepared the monthly, quarterly, and annual dose calculations for the public from plant operations. She reviewed the data from the environmental monitoring program to ensure that plant operations were not negatively impacting the environment. She was responsible for revising and maintaining the emergency plan and preparing emergency drill scenarios. During emergency drills, she directed the off-site teams as to where and what types of radiation, air, and environmental samples to collect.

PROFESSIONAL CHRONOLOGY: Radiation Monitor, Reynolds Electric & Engineering Co., Inc., 1979; Environmental Analyst, Reynolds Electric & Engineering Co., Inc., 1979-1980; Chemistry and Health Physics Technician, Vermont Yankee Nuclear Power Corp., 1980-1984; Environmental Coordinator, Vermont Yankee Nuclear Power Corp., 1984-1990; Environmental Scientist, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, 1990-1996; Health Physicist, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, 1996-present.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PRIVATE FUEL STORAGE L.L.C.	)	Docket No. 72-22-ISFSI
	)	
(Independent Spent Fuel Storage Installation)	)	
	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S RESPONSE TO APPLICANT'S MOTION FOR SUMMARY DISPOSITION OF UTAH CONTENTION C (DOSE LIMITS)" in the above captioned proceeding have been served on the following through deposit in the Nuclear Regulatory Commission's internal mail system, or by deposit in the United States mail, first class, as indicated by an asterisk, with copies by electronic mail as indicated, this 11th day of May, 1999:

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