July 2, 2009

Mr. Steven P. Kraft
Senior Director
Used Fuel Management
Nuclear Energy Institute
1776 I Street, NW
Suite 400
Washington, D.C. 20006-3708

SUBJECT: RESPONSE TO NEI'S FOLLOW-UP COMMENTS ON THE NRC/INDUSTRY MEETING OF NOVEMBER 21, 2008

Dear Mr. Kraft:

On November 21, 2008, the U.S. Nuclear Regulatory Commission (NRC), the Nuclear Energy Institute (NEI), and industry representatives met to discuss representative quality deficiencies and the level of detail necessary in the shielding and radiation protection chapters of the Safety Analysis Reports (SARs) for dry-cask storage and transportation systems. The Division of Spent Fuel Storage and Transportation (SFST) views meetings like this to be an important element of our continuous efforts to improve the licensing process. On March 4, 2009, you sent a letter to the NRC with comments and proposals on subjects discussed during the meeting.

Enclosure 1 to this letter responds to the six specific issues that were discussed in your letter. The second enclosure to this letter provides specific examples of application quality deficiencies as requested. Note that additional examples of application quality deficiencies can be found in NRC Regulatory Issue Summary 2007-09 (ADAMS Accession Number ML062550133), and in the requests for additional information that are issued during the review of specific applications.

As you are aware, the Division of Spent Fuel Storage and Transportation has been working on a number of initiatives designed to improve the licensing process. For example, draft NUREG-1536 Revision 1A, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," (ADAMS Accession Number ML090400676), and draft SFST Office Instruction 14, "Acceptance Review Process," (ADAMS Accession Number ML090960796), have been released for public comment and, when finalized, should improve the effectiveness and efficiency of our reviews. In addition, while not directly related to the meeting, SFST has been working on development of interim staff guidance (ISG) to establish a graded approach, based on projected dose rates, for performing radiation protection and shielding reviews for dry cask storage applications. We believe that this approach resolves your primary concern associated with the level of review effort in this area.

We appreciate the time and effort that NEI and the industry representatives provided to attend the meeting and to prepare the follow-up written response.

If you have any questions or additional comments, please do not hesitate to contact Michel Call or Jeremy Smith of my staff at (301) 492-3289 and (301) 492-3340, respectively.

Sincerely,

/RA/

Raymond Lorson, Deputy Director
Technical Review Directorate
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Enclosures: As stated

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Enclosures: As stated

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Comments Related to the Topics in NEI's letter of March 4, 2009

The Division of Spent Fuel Storage and Transportation (SFST) has reviewed the NEI March 4, 2009, letter and prepared responses. Although our responses generally focused on the requirements contained in Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," of the Code of Federal Regulations (10 CFR Part 72); certain responses considered additional 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," requirements. Additionally, note that our responses also considered the potential impact on our ability to conduct an effective review of the analytical methods submitted in support of a dry storage application. SFST considers the review of selected analytical methods to be an important element of our 10 CFR Part 72 review process since applicants often rely on these methods to perform system modifications.

Each of the numbered paragraphs below address the corresponding topic cited in the Enclosure to the NEI letter to SFST dated March 4, 2009.

1. Level of Detail in the Safety Analysis Report (SAR)

The requirements regarding the information to be included in a SAR are contained in 10 CFR Part 72 (e.g., 10 CFR 72.13, "Applicability"). During the review process, SFST strives to ensure that information to be included in the SAR is consistent with these requirements. While SFST understands there is an industry concern regarding the level of detail needed in the SAR, the requisite level of detail is case-specific; therefore, SFST is unable to generically agree with your proposal to limit the information contained in the SAR for all amendments for the reasons noted below. However, SFST is willing to review the requested level of detail for specific licensing actions. SFST agrees that the SAR shielding chapter should include the information listed in NEI's proposed resolution; however, there may be specific cases where information beyond that described in NEI's list is necessary. Thus, it is important for the applicant to communicate frequently with the licensing project manager during the review process to resolve any specific questions related to the level of detail to be included in the SAR.

The NEI letter also stated that there has been a significant increase in the level of detail requested to be placed in the SAR. While SFST seeks to maintain a consistent level of review and detail to be placed in the SAR, there appears to be a trend of increasing complexity and projected dose rates in dry storage applications (including amendments). Additionally, some applications (including amendments) have more closely approached design code acceptance limits, have changed or impacted the design basis, or have proposed new design features, changes to existing design features, significant changes to contents, and/or changes to analytic methods. For example, complex equations have been developed and proposed to define acceptable cask contents in some amendments. Also, in some amendments, spent fuel with burnup values significantly exceeding the validated range of the depletion codes used in the source term analysis have been proposed as allowable cask contents. SFST believes that the complexity of these newer designs, combined with the reduced design margins, has contributed to a greater need for information and detail than what may have been requested in previous reviews.

The NRC technical reviews are audit level reviews. To this extent confirmatory calculations will typically be performed on selected, or sometimes on a selected portion of a design calculation. If potential problems are discovered, then the review may be expanded. Generally, the staff practice is to request additional input files when the information provided in the SAR is not sufficient to perform an independent analysis. The level of effort expended during the staff's review is dependent on a number of factors including the complexity of the design and analysis, the estimated dose rates, safety margins, the codes used by the applicant, similarity to previous designs, and the anticipated use of the methodology to support potential future changes under 10 CFR 72.48. These factors will not only impact the extent of confirmatory reviews, but also the level of detail needed in the application. Historically, SFST has requested bounding sample input files to be included in the SAR. SFST may request additional input files to support the staff's review but does not typically require these additional files to be placed in the SAR.

SFST believes that providing technically rigorous dose estimates, and their associated bases in the SAR provides a significant safety benefit because radiation work planning efforts should begin well in advance of any spent fuel loading campaign and before any measured dose rate information is available. Reliance upon operating experience to project dose rates may not be sufficient due to differences in the source term, dry cask storage system, or site characteristics. In this case, information provided in the SAR (i.e., dose rates, operational exposure estimates) can form the basis for initial radiation work planning activities. Regardless of how the information in the SAR is used by plant personnel, the applicant still maintains the burden to demonstrate and document the acceptability of the proposed dry cask storage system as required by 10 CFR Part 72.

As discussed in the cover letter, SFST is developing an ISG that will define a graded approach to the level of staff review in the areas of shielding and radiation protection based upon projected dose rates. SFST believes that this ISG will address the concerns listed in your letter regarding the level of precision in applications.

2. Bounding Calculations and Results

SFST agrees with the comment that prior casework reviews should be utilized whenever possible to improve the efficiency associated with the review of new applications (including amendments). However, experience has shown that some follow-on amendments, that may appear similar to a previously approved application, contain new or unique features that impact the use of a prior review. For example, some include modifications that reduce shielding or expand the permissible basket storage locations for allowable contents that are currently restricted in an existing approved application. If the applicant is relying on the results of a prior review to support a new application (or amendment), the applicant must demonstrate that the new application (or amendment) is bounded by the prior design basis. This can require that key assumptions, limitations, uncertainty penalties, and supporting calculations be reviewed to ensure that the original design is bounding.

As discussed in the response to Issue 1, SFST performs an audit review of an application. As a result, it is possible that the staff may identify problems and issues that were not considered during the review of the original application. For example, SFST may perform a more robust review of a new application (including an amendment) that uses a methodology from a previously reviewed application if the new application (or amendment) has less margin. In addition, due to staff's review being an audit review, new issues may be identified that were not identified during an earlier review activity. The staff must ensure that any new issues or

concerns that are identified during the review of a new application (or amendment) are properly addressed for both the new and existing systems.

3. Definition of a Small Change versus a Small Effect

This is an area where the identified fuel characteristics may produce only a small effect for a 10 CFR Part 72 analysis but may be of concern for an application where the margin to the 10 CFR Part 71 dose rate limit is small. If the margin to the limit is small, an incorrect application of, or assumption regarding, any one of the fuel characteristics in the shielding analysis method may yield results that may incorrectly indicate compliance with the limit. For example, a transportation cask, for which calculated dose rates are within a few percent of a Part 71 limit, may exceed that limit when the impact of axial blankets of 8 to 12 inches on the fuel source term is included in the analysis, if not already considered. Concerns raised by the industry regarding this topic are a particularly good example of where the details related to a specific application (for a new design or amendment) should be discussed with SFST.

In general, SFST agrees that for 10 CFR Part 72 SARs, each of the identified inputs (or fuel characteristics) may have small effects on the independent spent fuel storage installation (ISFSI) dose rates (i.e., dose rates at the controlled area boundary), and only modest effects on dose rates on or near the transfer cask, storage cask, or horizontal storage module. When it is shown that fuel characteristics, such as those identified in NEI's letter, have little impact on demonstrating compliance with the regulatory requirements for a particular application, staff guidance is to not request excessive detail. However, SFST may review these parameters, if necessary, to determine the acceptability of a methodology discussed in a SAR.

4. Burnup and Cooling Time versus Heat Load Limits in the CoC

The values typically incorporated into the CoC and Technical Specifications are used to properly control the thermal and shielding performance of the cask and to provide information SFST can use to perform independent confirmatory calculations. Historically, decay heat limits have been used to define the acceptable bounding thermal source term, while burnup, cooling time and initial enrichment limits have been used to define the shielding source term. Further, for cask designs that are not leak tight in accordance with ANSI N14.5, or do not use the provisions contained in ISG-18, "The Design/Qualification of Final Closure Welds on Austentic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation," burnup, cooling time, and initial enrichment limits are used to define the acceptable limit for the confinement source (the amount of radioactive material available to be potentially released through the cask confinement boundary).

SFST has not seen any information or analysis to support a general conclusion that specifying parameters to control one of these three source terms (decay heat, shielding, or confinement) will adequately limit the others in all cases. For example, the heat load limits (decay heat source term) are calculated based on the burnup, cooling times, and initial enrichment. However, different burnups, cooling times, and enrichments may generate the same decay heat source term, but can result in different shielding source terms. Therefore, SFST is not able to address this proposal with a generic response. SFST is open to case-specific proposals where it can be demonstrated that only specifying the decay heat source term or just burnup, cooling time, and initial enrichment will provide adequate control over all three source terms and provide enough information for the staff to reach a proper regulatory determination.

As was the case for the previous topic, it should be emphasized that use of one set of parameter limits may be shown to be acceptable for a 10 CFR Part 72 application, but this may not necessarily be an acceptable approach for the associated 10 CFR Part 71 transportation system.

5. Technical Specification Dose Rates

Measuring the dose rate on the surface of the transfer cask (or canister, as appropriate) gives an early indication that the calculated dose rates are bounding. If the measured dose rates are not consistent with the calculations, this information can be used to revise radiological work planning activities. Further, since the radiation protection personnel at 10 CFR Part 50 facilities routinely monitor dose rates during fuel loading operations, assuring compliance with specified limits should not present an additional burden. In addition, dose rate measurements help confirm that there are no major shielding defects in the “as-built” condition (versus the “as-designed” condition), and provide assurance that critical system design parameters have been maintained.

SFST believes that misload detection is an additional benefit of establishing dose rate limits. The NEI letter states that they are unaware of the detection of any misloads by dose rate measurement; however, NRC Inspection Report 72-20/00-03 (ADAMS Accession Number ML010160098) highlights an instance where a misload was discovered by measurement. Specifically, unauthorized contents were loaded into a spent fuel canister, and this error was discovered by the measurement of unexpected neutron dose rate values. Thus, SFST believes that dose rate limits can be useful in the detection of misloads.

As stated in response to item 1, SFST is developing an ISG that will define a graded approach for determining the extent of the shielding and radiation protection reviews. It is envisioned that this ISG will establish appropriate limits to be included in the Technical Specifications.

6. Computer Codes

Computer codes used for shielding analyses do not necessarily need to be updated to the most recent version. However, the applicant assumes the burden to demonstrate that use of a code version, that is no longer supported by a vendor, is valid for the analysis, and also that the code has been properly maintained in accordance with the requirements contained in 10 CFR Part 72, “Subpart G-Quality Assurance.”

10 CFR 72.146, “Design control,” requires, in part, that measures must be established for the selection and review for suitability of application of processes that are essential to the functions of structures, systems, and components important to safety, and that these measures be applied to items such as shielding and radiation. Further, 10 CFR 72.150, “Instructions, procedures, and drawings,” requires, in part, that the applicant or certificate holder prescribe activities affecting quality by documented instructions or procedures and that these documents must include appropriate qualitative or quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished. When the developer terminates its support for a version of a code, it is difficult to establish that the code version used in an application has been properly maintained in accordance with the quality assurance (QA) requirements described above. Generally, SFST believes that supported codes are typically controlled by the developer in a manner consistent with 10 CFR Part 72, Subpart G requirements. Therefore, the applicant or certificate holder would assume the responsibility to ensure that the use of an unsupported code (version) is adequate for its intended application,

and that any identified errors, including those identified by other end users, have been properly evaluated and dispositioned prior to use in an application. Further, the regulations, previously cited, require that the applicant or certificate holder develop written procedures to ensure that codes no longer supported by the developer are adequate, continue to be error free, and prescribe a method for detecting and evaluating errors. Reliance on 10 CFR Part 21 reporting criteria would not necessarily provide assurance that all 10 CFR Part 72 quality assurance requirements were met.

Additionally, computer codes used for shielding analyses (including source term calculations) are developed and validated for a certain range(s) of parameters. Some of these codes may also have significant uncertainties or margins of error in their results due to the method that is the basis for the code. More recent codes are typically more rigorous in their method, thereby reducing the uncertainties in the results, and have expanded range(s) of parameter applicability for which the code is validated. Applicants should be aware of the area of validation and ensure that their use of the code is within this applicable range(s) of parameters. Use of codes outside their area of applicability introduces potential uncertainties that have not been addressed by the code developer and may lead to analytical errors and inaccurate conclusions that are not always conservative. These concerns may become more significant as modifications to cask systems and contents result in designs with higher dose rates. These concerns may increase in significance if the code used outside its area of validation is also a code that is no longer supported by its developer or vendor.

The comparison to Emergency Core Cooling System (ECCS) analyses is not applicable to codes used to perform shielding calculations. Unlike 10 CFR Part 50 Licensees, cask vendors adhering to 10 CFR Parts 71 and 72 are not specifically required to submit shielding codes for formal review and approval by SFST for each individual application. In addition cask vendors are not required to provide an annual report documenting the effect due to changes and errors found in the codes. In summary, SFST believes that it is appropriate to treat these software codes differently based on the different levels of review that each code receives.

SFST recognizes that there are some systems for which depletion codes have been used in applications significantly beyond their validated area of applicability. In these cases, penalties and other conservatisms, along with further limitations set forth in Technical Specifications, have been established and included in the method to attempt to address the uncertainties associated with these code uses. The reduction of these penalties and conservatisms or the modification of the added limitations can necessitate closer scrutiny by the reviewer and further justification by the applicant. When shielding codes and methodologies are described in an application for 10 CFR Part 72 activities, it is important to establish a well defined area of applicability in the SAR.

Common Areas of Deficiencies in Applications

The U.S. Nuclear Regulatory Commission (NRC) has compiled this partial list of common areas of deficiencies in applications to inform applicants and licensees of requests for additional information (RAIs) questions that have been asked when reviewing applications and amendments under 10 CFR Parts 71 and 72. These RAIs provide specific examples of shielding and radiation protection issues that were inadequately addressed in the Safety Analysis Report. This list is in addition to the information presented in RIS 2007-09, "Examples of Recurring Requests for Additional Information (RAIs) for 10 CFR Part 71 and 72 Applications," (ML062550133) and is targeted at shielding and radiation protection.

This listing has been divided into three overarching categories of deficiencies and provides general information on the issue without going into the specific requirements for each question.

1) Sufficiency of Information

a) ML080640507, RAI 11-1: No evaluation was provided of accident and natural phenomena events during excavation next to an operating ISFSI.

b) ML072980876, RAI 5-31: Additional information needed to describe the various shielding configurations and corresponding dose rates for normal and hypothetical accident conditions.

c) ML072980876, RAI 5-32: Several items requested to elaborate the shielding calculations.

d) ML083400083, RAI 10-2: A calculation was provided for a design feature without explanation of the basis and assumptions for the calculation.

e) ML070670418, Item 2: Sufficient justification for the proposed change and the bases for the change(s) in the conclusions that established a previous limit (the expansion of which is the focus of the proposed change) were not provided.

f) ML072080490, RAI 5-2: Neutron and gamma source terms for each fuel type were requested to account for differences in lattice configurations.

g) ML083010101, RAI 5-1: Demonstrate that the source-term for a new assembly was bounded by the design-basis assembly.

h) ML083010101, RAI 5-3: Justify the enrichment values provided in the application.

i) ML090540196, RAI 5-35: Applicant needed to revise the Technical Specifications to indicate that the dose rates to be included are those on the surface, at the controlled area boundary, and in the most affected unrestricted area.

j) ML090540196, RAI 5-37: Multipart RAI needed to clarify several items important to shielding.

k) ML0908400280, RAI A7.1: Sufficient information regarding burnup, enrichment, and cooling time combinations was not provided.

l) ML080570590, RAI 5-11: The SAR introduced a new equation without providing the technical basis or necessary reference(s).

m) ML072980876, RAI 5-49: Additional discussion on how worker doses were obtained was required.

2) **Regulatory Basis/Use of Approved Guidance**

a) ML080570590, RAI 5-7: The applicant cited a value that was derived from NUREG/CR-6801, however this document did not support their assertion.

b) ML071510035, RAI 5-2: Use of an older computer code and library that was not appropriate for the high burnup fuel in the applications.

c) ML083400083, RAI 11-1: The evaluation provided in response to a previous RAI (ML080640507, RAI 11-1) was incomplete, it only addressed a single accident event and neglected other important events. Also, the staff questioned the SAR provisions to defer evaluations of these other events to the end user and its 10 CFR 72.212 evaluation.

d) ML080570590, RAI 5-3: Inaccurate interpretation of the statements from NUREG/CR-6701 on the sensitivity study of burnup validation.

3) Internal Inconsistencies/Information Illegible

a) ML080640507, RAI G-1: The RAI points out inconsistencies within the SAR and between the SAR and the CoC and technical specifications.

b) ML080640507, RAI 1-1: The SAR figure showed features not discussed/evaluated in the SAR or shown in other drawings.

c) ML070670418, Item 5: The SAR did not address the change to the contents and configurations, and the impacts on the dose evaluations that are the subject of the item.

d) ML080570590, RAI 5-6: Pages in the SAR were incorrect.

e) ML080570590, RAI 5-11: An equation relating source term to fuel assembly burnup had numerous errors.

f) ML090540196, RAI 5-40: Various inconsistencies were identified.

g) ML090540196, RAI 5-42: Multiple inconsistencies noted in tables.

h) ML072080490, RAI 5-1: Needed to explain apparent discrepancies between the values for assembly thermal power and the values used in the calculations.

i) ML072080490, RAI 5-5: A discrepancy was identified in the type of cask bottom neutron shield material specified.

j) ML083400083, RAI G-1: The RAI requested proposed TS, CoC and a SAR consistent with the request for the storage contents. The base SAR document for the amendment did not include the analyses, and the TS did not contain the proposed design features

and contents drawn from a previously approved amendment that were used for the proposed changes in the amendment request.