November 9, 2006

Ms. Tara Neider
President and COO
Transnuclear, Inc.
7135 Minstrel Way, Ste. 300
Columbia, MD 21045

SUBJECT: NRC INSPECTION REPORT NO. 72-1004/2006-204 AND NOTICE OF VIOLATION

Dear Ms. Neider:

On August 28 - September 29, 2006, the U.S. Nuclear Regulatory Commission (NRC) staff performed an announced inspection of Transnuclear, Inc. (TN) at its office in Columbia, MD. The purpose of this inspection was to determine if TN's activities with regard to the design of the OS197L light weight transfer cask, conducted under the provisions of 10 CFR 72.48, were performed in accordance with the requirements of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-level Radioactive Waste, and Reactor-related Greater Than Class C Waste." The inspection scope included discussion and review of specific issues related to TN's 10 CFR 72.48 evaluation as documented in TN Evaluation Form LR No. 721004-321. The NRC staff had identified these issues during the pre-operational inspection, and subsequent exemption request, related to the independent spent fuel storage installation at Omaha Public Power District's Fort Calhoun site, which used the OS197L transfer cask during spent fuel loading operations. The enclosed report presents the findings from the inspection.

Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred. The violation is based on three examples wherein TN failed to seek a Certificate of Compliance (CoC) amendment when required by 10 CFR 72.48(c)(1). This violation was evaluated in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC's Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Enforcement Policy**. One severity level IV violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding the violation are described in detail in the subject inspection report. The violation is being cited because it was identified by the NRC. The team also identified several issues that would have constituted a violation of 10 CFR 72.48(c)(2) had the OS197L light weight transfer cask been used at Ft. Calhoun without the exemption that NRC had granted for its use at that facility. Lastly, the team identified weaknesses in TN's evaluation with regard to the technical bases in TN's Evaluation Form LR No. 721004-321 that caused TN to incorrectly conclude that use of the OS197L light weight transfer cask could be implemented without the need for a CoC amendment.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. For your consideration and convenience, an excerpt from the NRC Information Notice 96-28, "Suggested Guidance Relating to Development and Implementation of Corrective Action," is enclosed. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Documents Access and Management System (ADAMS) accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction.

Sincerely,

/RA/

Robert J. Lewis, Chief
Rules, Inspection and Operations Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 72-1004

Enclosures:
1. NRC Inspection Report No. 72-1004/2006-204
2. Notice of Violation
3. 10 CFR 72.48 Issues for Fort Calhoun Inspection, August 3, 2006
4. Excerpt from NRC Information Notice 96-28, "Suggested Guidance Relating to Development and Implementation of Corrective Action"

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U.S. NUCLEAR REGULATORY COMMISSION
Office of Nuclear Material Safety and Safeguards
Division of Spent Fuel Storage and Transportation

Inspection Report

Docket:	72-1004
Report:	72-1004/2006-204
Certificate Holder:	Transnuclear, Inc. 7135 Minstrel Way, Ste. 300 Columbia, MD 21045
	Date: August 28-29 and September 29, 2006
Inspection Team:	James Pearson, Team Leader, SFST Joseph Sebrosky, Senior Project Manager, SFST Shana Helton, Nuclear Engineer, SFST Jorge Solis, Nuclear Engineer, SFST
Approved by:	Robert J. Lewis, Chief Rules, Inspection and Operations Branch Division of Spent Fuel Storage and Transportation Office of Nuclear Material Safety and Safeguards

Enclosure 1

EXECUTIVE SUMMARY

Transnuclear, Inc.
NRC Inspection Report 72-1004/2006-204

On August 28 - 29, 2006, the U.S. Nuclear Regulatory Commission (NRC) staff performed an announced inspection of Transnuclear, Inc. (TN) at its office in Columbia, MD. The purpose of this inspection was to determine if TN's activities with regard to the design of the OS197L light weight transfer cask (designated by TN as the OS197L TC), conducted under the provisions of 10 CFR 72.48, were performed in accordance with the requirements of 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-level Radioactive Waste, and Reactor-related Greater Than Class C Waste." The inspection scope included discussion and review of specific issues related to TN's 10 CFR 72.48 evaluation as documented in TN Evaluation Form LR No. 721004-321. The NRC staff had identified these issues during the pre-operational inspection, and subsequent exemption request, related to the independent spent fuel storage installation at the Omaha Public Power District (OPPD) Fort Calhoun site, which used the OS197L TC during spent fuel loading operations.

Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred. The Level IV Violation is based on three examples wherein TN failed to seek a Certificate of Compliance (CoC) amendment when required by 10 CFR 72.48(c)(1). The team also identified several issues that would have constituted a violation of 10 CFR 72.48(c)(2) had the OS197L TC been used at Ft. Calhoun without the exemption that NRC had granted for its use at that facility. Lastly, the team identified numerous weaknesses in TN's evaluation with regard to the technical bases in TN's Evaluation Form LR No. 721004-321 that caused TN to incorrectly conclude that use of the OS197L TC could be implemented without the need for a CoC amendment.

Enclosure 2 contains the Notice of Violation associated with this report and Enclosure 3 contains a summary of the issues on which the team focused during this inspection. The information in Enclosure 3 was provided to TN prior to the inspection so that they would be aware of, and be prepared to address, these issues during the inspection. The items contained in Enclosure 3 were identified as a result of an NRC inspection that was performed at Fort Calhoun [see Inspection Report 050-00285/06-17; 072-0054/06-003, dated August 28, 2006, (ADAMS Accession Number ML062410221)].

REPORT DETAILS

1. Background Information

<u>Inspection background</u>

TN prepared an evaluation, pursuant to 10 CFR 72.48 and their NRC approved quality assurance program, identified as LR No. 721004-321, Revision 1, to add the OS197L TC to the generally-licensed Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel. OPPD's Fort Calhoun facility intended to incorporate TN's 72.48 through their 72.212 evaluation in order to use the OS197L TC for spent fuel loading operations. The NRC staff identified several issues associated with TN's application of the 10 CFR 72.48 process during the pre-operational inspection at OPPD's Fort Calhoun (FC) site. Interactions with the NRC staff led OPPD management to the determination that submittal of an exemption request was the optimal path forward for use of the OS197L TC at FC. On August 28 - 29, 2006, SFST staff performed an inspection at TN's Columbia, MD, office to discuss the previously-identified issues associated with the 72.48 developed by TN for the use of the OS197L TC. A final exit meeting, by conference call, was held between the NRC and TN personnel on September 29, 2006.

<u>OS 197L TC Background</u>

TN used the 72.48 process to evaluate the creation and use of a light weight transfer cask design designated as the OS197L TC. TN indicated in their 72.48 analysis that the OS197L TC was developed in an effort to expand the capability of the NUHOMS® system for plants with reduced crane capacity. The OS197L TC has reduced shielding, including the elimination of all the lead shielding from previous versions of the TC, and results in a lower weight for the TC. The TC that the OS197L TC replaces (designated as the OS197 TC) requires a 100 ton crane capacity. Because the OS197L TC has less shielding (including the elimination of all the lead shielding) than the OS197 TC, the OS197L TC surface dose rates are higher than the OS197 TC with lead shielding. To reduce personnel doses, crane operations associated with the

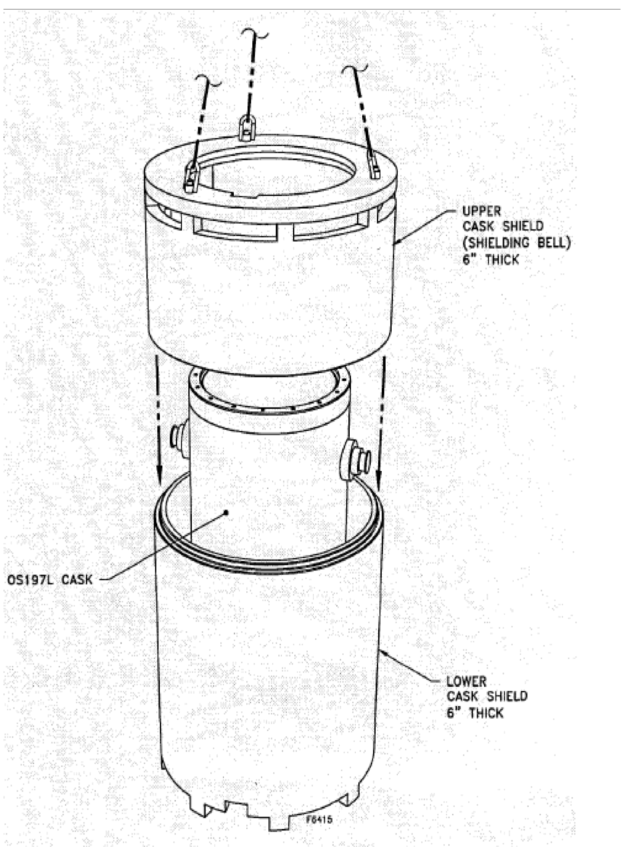

Figure 1 - Supplemental Shielding in Decontamination Area

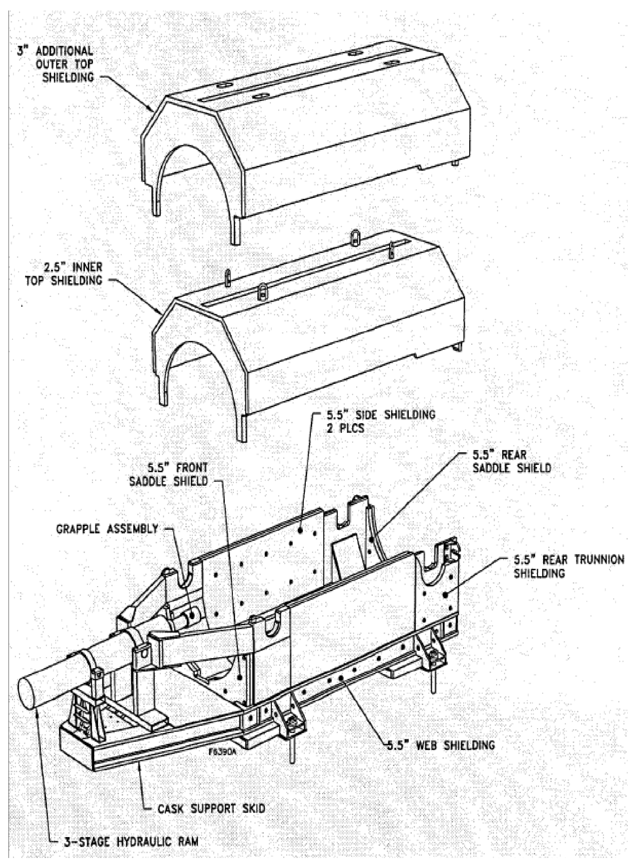

Figure 2 - Supplemental Shielding on Transfer Trailer

OS197L TC are done remotely and supplemental shielding is provided in the decontamination area where the dry shielded canister (DSC) is welded, and on the transfer trailer that is used to transport the OS197L TC to the horizontal storage module (HSM). The supplemental shielding used in the decontamination area is shown in Figure 1. The supplemental shielding provided on the transfer trailer is shown in Figure 2.

2. Inspection Process for Review of TN's 72.48 for the OS197L TC

Inspection Scope and Purpose

The purpose of the inspection was to assess the effectiveness of TN's performance of 10 CFR 72.48 evaluations for use of the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, and to assess the compliance of those evaluations and the evaluation process with 10 CFR 72.48 and TN's NRC-approved quality assurance program. The inspection consisted of a detailed examination of TN's 10 CFR 72.48 evaluation LR No. 721004-321, Revision 1, "10 CFR 72.48 Applicability and 10 CFR 71 Review Form." The inspection was focused on issues specific to this evaluation performed by TN for the OS197L TC. The staff had identified these issues during the pre-operational inspection and subsequent exemption request related to the independent spent fuel storage installation (ISFSI) at OPPD's FC site, which used the OS197L TC during spent fuel loading operations.

The staff's activities related to the FC ISFSI consisted of audits of selected OPPD and TN documentation supporting OPPD's 72.212 evaluations, and interviews with cognizant FC, TN, and contractor personnel. OPPD's 72.212 evaluation referenced TN's 72.48 analysis performed to incorporate the OS197L TC into the Standardized NUHOMS system. As a result of the issues identified with TN's 72.48 analysis during the FC pre-operational inspection, and consequent exemption activities, the staff performed a reactive inspection of TN's 10 CFR 72.48 evaluation for the OS197L TC. The inspection team audited selected supporting documentation and calculation files to clarify or confirm the bases and methods used by TN in their evaluation. The accuracy of the calculations was not evaluated by the team. Following the document review, the team performed an on-site inspection at TN's office in Columbia, MD.

The team determined that TN used Nuclear Energy Institute's (NEI) NEI 96-07, Appendix B, "Guidelines for 10 CFR 72.48 Implementation," dated March 5, 2001, for guidance in implementing the provisions of 10 CFR 72.48. Appendix B was endorsed by the NRC in Regulatory Guide (RG) 3.72, "Guidance for Implementation of 10 CFR 72.48, Changes, Tests, and Experiments," dated March 2001, and provided methods that were acceptable to the NRC staff for complying with the provisions of 10 CFR 72.48.

Findings and Observations

As described in the following report Sections, the inspection team identified the following issues:

- A Level IV Violation, based on three examples, wherein TN failed to seek a Certificate of Compliance (CoC) amendment when required by 10 CFR 72.48(c)(1).

- Several issues that would have constituted a violation of 10 CFR 72.48(c)(2) had the OS197L TC been used at OPPD's FC station without the exemption that NRC had granted for its use at that facility.

- Weaknesses in TN's evaluation with regard to the technical bases in TN's Evaluation Form LR No. 721004-321 that caused TN to incorrectly conclude that use of the OS197L TC could be implemented without the need for a CoC amendment.

3. Violation of 10 CFR 72.48(c)(1)

The team identified a violation of the requirements of 10 CFR 72.48(c)(1), which state, in part, that a certificate holder may make changes in the spent fuel storage cask design as described in the Final Safety Analysis Report (FSAR), and make changes in the procedures as described in the FSAR, without obtaining a CoC amendment, if a change in the terms, conditions, or specifications incorporated in the CoC is not required. Contrary to this requirement, the inspection team identified the following three examples where TN failed to obtain a CoC amendment for changes made in the spent fuel storage cask design and procedures, as described in the FSAR that resulted in a change in the specifications (i.e., technical specifications) incorporated in the CoC.

As discussed further in Sections 4 and 5 of this Enclosure, the team identified weaknesses in TN's evaluations that caused TN to incorrectly conclude that use of the OS197L TC could be implemented without the need for a CoC amendment.

A. Issue Related to Technical Specification (TS) 1.2.17a

For TS 1.2.17a, "32PT DSC Vacuum Drying Duration Limit," the Limits/Specification section states the following:

1. The limit for duration of Vacuum Drying is 31 hrs for a 32PT DSC with a heat load greater than 8.4 kW and up to 24 kW after initiation of vacuum drying.

2. The limit for duration of Vacuum Drying is 36 hrs for a 32PT DSC with a heat load of up to 8.4 kW after initiation of vacuum drying.

The objective of this TS states:

> To ensure the fuel cladding temperature in the 32PT DSC does not exceed 752°F during drying and also to meet the thermal cycling limit of 117°F during drying, helium backfilling and transfer operations.

Section M.4.7.1 of the FSAR revision 8, "Maximum Fuel Cladding Temperatures During Vacuum Drying," for the Standardized NUHOMS® design states, in part, that: "The loading condition evaluated for the NUHOMS® 32PT DSC is the heatup of the DSC before its cavity is backfilled with helium... A transient analysis is performed using the three-dimensional model developed in Section M.4.4.1, decay heat loads for heat load zoning configuration 1, 2, and 3 and a maximum DSC temperature of 215°F. The initial temperature of the DSC, basket and fuel is assumed to be 215°F, based on the boiling temperature of the fill water."

The team determined that the FSAR, Section M.4.7.1 for the Standardized NUHOMS® design, describes a methodology for determining the initial value for the temperature of the basket and fuel cladding at the start of vacuum drying. The discussion in Section M.4.7.1 is consistent with the sequence of operations described in FSAR Section M.8.1.3. This section of the FSAR discusses the DSC drying and backfilling process and that the inner top cover plate welding operation is done with water in the DSC. After the welding is completed and the dye penetrant weld examination is performed in accordance with TS 1.2.5, pump down of the cask is then performed.

In their evaluation for the OS197L TC, TN modified the procedure sequence in the FSAR, Section M.8.1.3, to allow for the water in the DSC to be pumped down much earlier in the process. The team's position is that this change in the procedural sequence for water pump down adversely affects the assumed initial cladding temperature used to calculate the TS allowed vacuum drying times. Specifically, TN's assumption in their evaluation that the 215° F cladding temperature is met at the beginning of vacuum drying is not conservative. The team considers that the initial fuel cladding temperature, at the start of vacuum drying in TN's revised pump down sequence, could be higher than the FSAR assumed value of 215° F because of the amount of time that the bulk of the water from the DSC is removed, when compared to that described in the FSAR. An assumed temperature of the fuel cladding higher than the 215°F basis in the FSAR would result in a shorter vacuum drying time than that specified in TS 1.2.17a. Therefore, the team's position is that TN should have sought a CoC amendment to reflect the shorter vacuum drying time limits that would result from the higher initial cladding temperature.

TN's failure to obtain a CoC amendment to change the TS 1.2.17a values to recognize the OS197L TC revised procedural sequence for water pumpdown is Example 1 of the Violation cited in Enclosure 2 of this inspection report.

B. Issues Related to TS 1.2.11

TS 1.2.11, "Transfer Cask Dose Rates with a Loaded 24P, 52B, 61BT, or 32PT DSC," has the following Limits/Specifications:

> Dose rates from the transfer cask shall be limited to levels, which are less than or equal to:
>
> a. 200 mrem/hr at 3 feet with water in the DSC cavity.
>
> b. 500 mrem/hr at 3 feet without water in the DSC cavity

The objective of the TS states:

> The dose rate is limited to this value to ensure that the DSC has not been inadvertently loaded with fuel not meeting the specifications in Section 1.2.1 of the TS, and to maintain dose rates as low as is reasonably achievable during DSC transfer operations.

The TS bases are the shielding analyses presented in Section 7.0, Appendix J, Appendix K, and Appendix M of the FSAR.

TN eliminated the lead shielding, and otherwise modified the OS197L TC design, to make the TC lighter. To compensate for the reduction of the shielding in the TC, TN provided additional temporary shielding in the decontamination area and on the transfer trailer. The additional temporary shielding is not present when the TC is lifted from the pool and moved to the decontamination area or when it is lifted from the decontamination area and placed on the trailer. TN's 72.48 analysis states, in part that the OS197L TC supplemental shielding, in conjunction with the OS197L bare cask body, provides a level of shielding that is similar to the OS197 TC.

TN provided a table of OS197L TC Normal Condition Dose Rates in LR 721004-321, Section 5.1.3. TN calculated the total surface dose rate for a normal weight OS197 TC to be 346 mrem/hour, a value that is well under the TS limits of 200 and 500 mrem/hr measured at 3 feet from the TC. However, for the OS197L TC, TN calculated the surface dose rate to be 53,249 mrem/hr. This value is over 100 times higher than that calculated for a normal weight OS197 TC, and as a result, the TS limits of 200 and 500 mrem/hr at 3 feet are essentially rendered meaningless as they would be exceeded from the very start, and more importantly, they no longer provide relevant values for meeting the TS objectives, which are to ensure that the DSC has not been inadvertently loaded with fuel not meeting the specifications in Section 1.2.1 of the TS, and to maintain dose rates as low as is reasonably achievable (or ALARA). Because of this, TN should have requested an amendment to the CoC to provide meaningful TS 1.2.11 values.

One reason that TN did not seek a CoC amendment was that the OS197L TC was designed to be used with temporary shielding during certain, but not all, phases of TC movement. With the additional temporary shielding, the surface dose rate at the temporary shielding surface is calculated to be 122 mrem/hr. TN's 72.48 analysis stated, in part, that the OS197L TC supplemental shielding, in conjunction with the OS197L TC bare cask body, provides a level of shielding that is similar to the OS197 TC. However, the team determined that TS 1.2.11, *as written*, requires dose rates to be measured at a distance of 3 feet from the TC, not 3 feet from the temporary shielding around the TC . TS 1.2.11 does not recognize the use of temporary shielding with respect to the dose rate limits of 200 and 500 mrem/hr at 3 feet. As such, the dose rate values in TS 1.2.11 are not applicable to the OS197L TC configuration. Therefore, TN should have requested a CoC amendment to change the TS 1.2.11 values to recognize the OS197L TC configuration.

TN's failure to obtain a CoC amendment to change the TS 1.2.11 values to recognize the OS197L TC configuration is Example 2 of the Violation cited in Enclosure 2 of this inspection report.

C. Issues Related to TS 1.2.1

TS 1.2.1, "Fuel Specification" states, in part, in its objective: "This specification is prepared to ensure that the ... maximum surface doses ... are below the design limits." The TS states the following in its bases: "The radiological design criterion is that fuel stored in the NUHOMS® system must not increase the average calculated HSM or TC surface dose rates beyond those calculated for the 24P, 24PHB, 52B, 61BT, or 32PT canister full of design basis fuel assemblies with or without BPRAs. The design value average HSM and cask surface dose rates for the 24P and 52B canisters were calculated to be 48.6 mrem/hr and 591.8 mrem/hr respectively based on storing twenty four (24) Babcock and Wilcox 15x15 PWR assemblies (without BPRAs) with 4.0 wt. % U-235 initial enrichment, irradiated to 40,000 MWd/MTU, and having a post irradiation time of five years." TN's 72.48 analysis (LR721004-321) states that the OS197L TC is designed to allow for the loading/unloading and transfer of the 24P, 52B, 61BT, 24PT2, 32PT and 24PHB DSCs. Therefore, the 24P and 52B DSCs are within the scope of TN's 72.48.

With regard to the NUHOMS CoC 1004, it is the team's position, in this particular case, that the TS bases are part of the TS and that they cannot be changed without prior NRC approval. For TS 1.2.1, the bases are written as part of the TS and are included in the attachment to the CoC 1004. The bases for TS 1.2.1 state that the radiological design criterion is that fuel stored in the NUHOMS system must not increase the average calculated TC surface dose rates beyond those calculated previously. It further states that the TC surface dose rates for the 24P and 52B are calculated to be 598.1 mrem/hr. As noted in Section 3.B above, TN calculated a TC surface dose rate that is two orders of magnitude higher than the 598.1 mrem/hr limit. The team noted that neither the TS nor its basis refer to the use of temporary shielding in meeting these values. Therefore, TN should have requested a CoC amendment to change the TS 1.2.1 values to recognize the OS197L TC configuration.

TN's failure to obtain a CoC amendment to change the TS 1.2.1 values to recognize the OS197L TC configuration is Example 3 of the Violation cited in Enclosure 2 of this inspection report.

It should be noted that although the above discussion involves dose rate values associated with the 24P and 52B DSCs, in the exemption granted to OPPD, the staff concluded that an exemption from the following text in the bases of TS 1.2.1 was also necessary: "The radiological design criterion is that fuel stored in the NUHOMS® system must not increase the average calculated HSM or transfer cask surface dose rates beyond those calculated for the 24P, 24PHB, 52B, 61BT, or 32PT canister full of design basis fuel assemblies with or without BPRAs."

The staff further stated in the exemption that: "Due to the reduced shielding of the OS197L TC, the TC surface dose rates will increase beyond those calculated for the 32PT design basis fuel when the OS197L TC is used for fuel-transfer operations. While specific dose rate limits for the 32PT are not given in this bases section, the TS was written for a TC with different, more robust shielding characteristics. The surface dose rate of the OS197 TC (more robust shielding than the OS197L TC) loaded with the 32PT DSC with design basis fuel in Table M.5-5 of the FSAR is 950 mrem/hour. The surface dose rate of the OS197L TC loaded with the 32PT DSC with design basis fuel (not the fuel permitted in this exemption) was calculated to be 53,000 mrem/hour (56 times higher than the FSAR Table M.5-5 value). The staff does not accept the view that the surface dose rates can be calculated on the supplemental shielding rather than on the surface of the TC."

Ths staff also stated that "Changing the design basis fuel and reducing the shielding in the TC have the same effect of increasing TC surface dose rates. The bases for this TS clearly indicate that the use of different fuel must not increase the average calculated TC surface dose rates. The staff takes this to mean that one of the primary goals of this TS is to ensure that the TC surface dose rate does not significantly increase. Use of the OS197L TC significantly increases the TC surface dose rate, hence the staff believes an exemption is needed from this portion of the bases as well."

4. Issues Related to 10 CFR 72.48(c)(2)

As a result of the team's review of TN's 72.48 for the OS197L TC, an issue was identified with regard to 10 CFR 72.48(c)(2) requirements. Specifically, had use of the OS197L TC been implemented at Ft. Calhoun without the exemption that OPPD sought for its use, then a violation of 72.48(c)(2) requirements could have been issued. This is because TN's 72.48 evaluation for the OS197L TC that OPPD endorsed through its 72.212(b) evaluation, contained unapproved departures from methods of evaluation described in the FSAR. As used in this inspection report and in 72.48(c)(2), the term "implementation" is defined in Section B4.5 of NEI 96-07, Appendix B (as endorsed by the NRC in RG 3.72) as follows: "An activity is considered "implemented" when it provides its intended function, that is, when it is placed in service and declared operable."

The regulations in 10 CFR 72.48(c)(2) state, in part, that "... a certificate holder shall obtain a CoC amendment pursuant to 10 CFR 72.244, and a general licensee shall request that the certificate holder obtain a CoC amendment pursuant to 10 CFR 72.244, prior to implementing a proposed change, if the change would...(viii) Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses." During its review of TN's 72.48 for the OS197L TC, the team identified the following examples where TN's evaluation included departures from methods of evaluation described in the FSAR used in establishing the design basis or in the safety analyses. The team also assessed that there were weaknesses in TN's evaluation that contributed to TN's failure to identify that changes in methodology were involved.

It should be noted that for the two examples discussed below, the exemption that NRC granted to OPPD obviated the need for OPPD or TN to address these issues prior to implementation of the OS197L TC.

A. Change in Methodology Associated with Use of FLUENT Code

TN used the FLUENT Code Computational Fluid Dynamics (CFD) methodology to define boundary condition temperatures for the OS197L TC inside the supplemental shielding on the transfer trailer. Figure 2 in Section 1 of this Enclosure shows the transfer trailer supplemental shielding. The TC is placed inside this shielding when it is moved from the spent fuel pool building to the horizontal storage module. TN's position is that use of the FLUENT code is appropriate to model the natural convection driven air flow inside the shielding as it flows around the TC. The CFD analysis is performed modeling the cask and skid geometry including the air flow paths in the cask/skid shielding enclosure. The shielding enclosure is provided with openings between the skid beams and the trailer deck to allow air to enter the enclosure. Air exits the enclosure through an opening at the top of the enclosure.

TN's position is that the overall geometry of an enclosure with rectangular geometry from the approximate centerline of the cask to the bottom of the enclosure and a curved/truncated octagonal configuration above the cask centerline is similar in size, configuration and geometry to the NUHOMS® Advanced Horizontal Storage Module (AHSM). TN used the FLUENT Code CFD thermal analysis associated with the AHSM to evaluate the TC inside the transfer trailer shielding.

During the inspection, TN informed the team that TN's reasoning for not changing the method of evaluation for the thermal model of the TC inside the transport trailer was based on the following:

1. LR 721004-321 Form 3.5-3 page 19-22 documents the methodology used and concludes that there is not a change in method of evaluation,

2. the safety evaluation report for the AHSM (docket 72-1029) accepted FLUENT without constraints including natural circulation within a system,

3. ICEPACK and GAMBIT are essentially the same and are "mesh generators" that create the computational space and it does not constitute a methodology,

4. the 24PTH SER amendment 8 approved the extrapolation method used by TN, although TN conceded that this is not mentioned in their 72.48 analysis,

5. that the AHSM methodology is for a cylinder inside an enclosure and that this methodology was approved for this intended application since both involve convective air flow and have similar heat transfer properties, and that the intended application is the same as modeling the transfer cask inside the transport trailer temporary shielding,

6. that TN has not changed methodology, only input parameters, and

7. that TN had made these determinations with the help of NEI 96-07, Sections B.4.3.8.2, B.3.8, and B.4.3.8.1.

The team, however, does not agree with TN's position that the AHSM methodology could be used to model a TC inside the transport trailer shielding. The team's position is that modeling a DSC inside an AHSM is different than modeling a DSC inside a transfer cask inside an AHSM. The team considers that the appropriate guidance in this case is contained in NEI 96-07, Appendix B, page 67, that states: "Similarly, licensees or cask certificate holders can also use different methods without first obtaining a license or cask CoC amendment if those methods have been approved by the NRC for the intended application." The team does not consider that the AHSM methodology was intended for the application that TN used it for; i.e., modeling a DSC inside a TC inside an HSM. The team also does not consider the DSC inside the TC inside the HSM to be adequately described as a cylinder within an enclosure, in that it is more accurately described as a cylinder, inside a cylinder filled with water, inside an enclosure.

The team considers that there are several geometry differences and modeling differences that led to the team's conclusion that TN's evaluation involved a change in method of evaluation. The geometry of the OS197L TC inside the supplemental shielding is different than the geometry of a DSC inside an AHSM. Differences between the geometries and how they affect the thermal methodology are listed in Table 1 below.

Table 1 - Geometry Differences

Item # and description	**DSC Inside AHSM**	**DSC Inside OS197L TC Inside Transfer Trailer Supplemental Shielding**
1a) Modeling of Impediment to Heat Transfer Posed by Transfer Cask	Not Done - Transfer cask is never inside the AHSM	TN uses simplified calculations to determine temperature profile on the outside of the OS197L TC and then extrapolates these temperatures back to the fuel cladding inside the DSC. The staff has never approved such a methodology.
1b) Modeling of Temperature Profile on the Outside of the DSC or TC	Assumes the temperature is uniform around the outside of the DSC	Assumes the temperature is uniform on the outside of the TC. The team does not agree with this assumption and considers that the temperature distribution is much more complex given the annular air gap and movement of the water in the neutron shield.
1c) 3D modeling	3D modeling done of entire length (approx. 190 inches) of AHSM	Essentially a 2D model. Cross section model done for 12 inch section of OS197L TC inside trailer supplemental shielding instead of for the entire length of the DSC.

In addition to the basic geometry differences, the team also identified differences in the way that TN modeled the heat transfer process. These differences between the heat transfer mechanisms and how they affect the thermal methodology are shown below in Table 2.

Table 2 - Thermal Modeling Differences

Item # and Heat Transfer Mechanism	**DSC Inside AHSM**	**DSC Inside OS197L Inside Transfer Trailer Supplemental Shielding**
2a) Radiation Heat Transfer	Based on View Factor (Surface to Surface). In this model radiation, heat transfer from the concrete to the environment is small compared to the heat transfer from the movement of the air.	Based on discrete ordinate methodology. For this configuration, radiation is an important contributor when it comes to heat rejection capabilities. The supplemental steel shielding gets hot in the model and rejects heat via this mechanism.

Item # and Heat Transfer Mechanism	DSC Inside AHSM	DSC Inside OS197L Inside Transfer Trailer Supplemental Shielding
2b) Environment	Not Modeled - Boundary conditions are applied on the AHSM external concrete walls (i.e., the thermal model ends at the exterior concrete wall for the AHSM).	Modeled Explicitly - heat transfer and flow analysis is performed on the supplemental shielding external walls (i.e., the model does not end at the exterior shielding surface). TN extends the model past this point to model air movement and the resultant heat transfer. The staff has never previously approved a model that goes beyond the exterior concrete surface of the HSM.
2c) Codes Used	CFD analysis is based on Fluent/Icepak Codes	CFD based on Fluent and Gambit Codes. These codes have not been properly validated or previously approved for the intended function for which TN is using them. The meshing capabilities of Icepak and Gambit are different. Icepak can provide only for uniform mesh. Gambit can be used to generate non-uniform mesh especially close to a heated wall. Obtained thermal results will depend on the type of mesh being utilized. The effect that these two meshing approaches (uniform versus non-uniform) has on the predicted results was not evaluated in TN's 72.48 analysis.

During the inspection, TN stated that the 24PTH safety evaluation report (SER), amendment 8, approved methods to calculate TC component temperatures including the estimates of peak cladding temperature. However, TN conceded that this was not specifically mentioned in their 72.48 analysis. The team reviewed the 24PTH, amendment 8, SER and the safety analysis report (SAR) for the 24PTH and could not conclude that the evaluation method that TN provided in its 72.48 evaluation was the same method described in the 24PTH SAR and SER.

The team reviewed TN's calculation NUH06L-0401, "CFD Evaluation of OS197L and OS197L100 Transfer Casks within Transfer Skid Auxiliary Shielding," dated March 9, 2006. The team noted in its review of this calculation that the supplemental shielding around the TC acts as an impediment to heat transfer. Based on the calculation, there was an increase of 44°F in the average TC outer surface temperature. This 44°F increase resulted in an increase of approximately 10°F in the maximum fuel cladding temperature from 720°F to 730°F. The team expected that the peak cladding temperature for the fuel would have been higher than 10°F.

NEI 96-07, Appendix B, page 69, states: "Changes to elements of analysis methods that yield conservative results, or results that are essentially the same over the entire range of use for the method would not be departures from approved methods." The team concluded that the geometry and thermal modeling differences mentioned above were not properly evaluated and documented over the entire range of use and thus involved a change in method of evaluation for which TN should have sought a CoC amendment under the provisions of 10 CFR 72.48(c)(2). Furthermore, based on the results of TN's calculations the staff believes that TN's calculations have underestimated the effect the additional shielding on the transfer trailer would have on the peak cladding temperatures of the fuel.

B. Change in Methodology Associated with TS 1.2.17a

With regard to TS 1.2.17a issues discussed in Section 3.A of this Enclosure, the team concluded that TN changed the methodology for the thermal evaluation described in the FSAR associated with the maximum fuel cladding temperature during vacuum drying. During the inspection, TN stated that their basis for concluding that a change in method was not involved included the following:

1. it was not considered a change in method of evaluation, rather it was a change in an input parameter that is allowed to be made under 72.48,

2. other designs like the 61BT and the 32PTH have different initial temperatures of the basket and fuel cladding which provide examples of engineering judgement to determine an input parameter,

3. the 215°F initial temperature is still a valid initial temperature for the fuel cladding given the amount of time it would take to heat the fuel cladding from the 80-100°F range to the 215°F range (TN verbally stated that they had performed a back-of-the-envelope calculation showing an approximate cask heat-up rate of 4-10°F per hour; however, TN stated that this heat-up rate calculation had not gone through their quality assurance program),

4. LR 721004-321 Form 3.5-1 page 25 of 26 states no change to TS 1.2.17a is required because there are identical boundary conditions,

5. LR 721004-321 Form 3.5-3 page 18 of 22 evaluates departures from change in method of evaluation in the thermal area, and

6. TN contended that there was no TS providing a time limit for bulk water removal. TN's verbal citation for why this change could be done under 72.48 was Section B.3.8, and B.3.10 of NEI 96-07, Appendix B, which discusses changes that can be made to input parameters.

The team's position is that changing the sequence of operations such that the bulk of the water in the DSC is removed prior to the TC leaving the spent fuel pool changed the assumptions used to support TS 1.2.17a and that this constituted a change in method of evaluation. The team based this conclusion on guidance contained in NEI 96-07, Appendix B, section B.3.8, page 22 which states, in part, that: "On the other hand, an input parameter is considered to be an element of the methodology if: the method of evaluation includes a methodology describing how to select the value of an input parameter to yield adequately conservative results." Additionally, Section B.3.10 states that dose conversion factors and assumed source terms are elements of a methodology. The team concluded that changing the water level in the DSC, which in turn may affect the initial temperature of the fuel cladding used in the vacuum drying analysis, is the same as changing a source term and therefore is an element of a methodology.

The team considers that the discussion in the FSAR, Section M.4.7.1, for the Standardized NUHOMS® design describes a methodology for determining the initial value for the temperature of the basket and fuel cladding at the start of vacuum drying. The discussion in FSAR Section M.4.7.1 is consistent with the sequence of operations described in the FSAR, Section M.8.1.3. This Section of the FSAR discusses the DSC drying and backfilling process and that the inner top cover plate welding operation is done with water in the DSC. After the welding is completed and the dye penetrant weld examination is performed in accordance with TS 1.2.5, pumpdown of the cask is then performed. When the sequence of operations was changed so that the water in the DSC was pumped down much earlier in the process, it impacted this assumption in the FSAR. As the team considers that the DSC water level was an element of a methodology, then TN's evaluation that changed the pumpdown sequence involved a change in methodology that required prior NRC approval per TS 72.48(C)(2)(viii) prior to implementation.

5. Weaknesses in TN's Evaluations and Technical Bases

As a result of the team's review of TN's 72.48, a number of weaknesses were noted with regard to TN's evaluation and the technical bases that incorrectly led TN to conclude that the OS197L TC could be implemented without requesting a CoC amendment from the NRC. In addition to weaknesses noted in TN's evaluations regarding changes in methodology, as discussed in Section 4 above, the following weaknesses in TN's evaluations were also identified by the team:

A. Weaknesses in Documentation of Change in Shielding Methodology

TN's 72.48 shielding evaluation in LR No. 721004-321, Revision 1, Form 3.5-3 section 5.1.3 referenced calculation NUH06L-0501, Rev. 1. Calculation NUHO6L-0501, Rev. 1, presented a methodology using simple extrapolation to evaluate the loss of the inner skin of the neutron shield of 0.45 inches of steel. Based on its audit of this calculation, the team concluded that TN's documentation was weak with regard to the conclusion that the evaluation did not involve a change in method of evaluation.

The team assessed that the simple calculation contained in NUH06L-0501, Rev.1, which uses a ratio of dose rates, is likely not conservative. Specifically, the hand calculation is a one-dimensional calculation that does not account for buildup (i.e., secondary scattering of radiation in material) nor does it take into account the change in spectrum of the gammas due to the loss of lead shielding. This simplification can lead to an under prediction of the dose rates and should not generally be used for anything but very minor changes in the shielding configuration or source term.

Chapter M.5.4 of the Standardized NUHOMS FSAR describes the shielding calculations that were performed to evaluate the loss of the inner skin of the neutron shield (0.45 inches of steel) for the 32PT DSC and its associated TC during accident conditions. The methodologies described in Chapter M.5.4.5 of the FSAR, and used for the bulk of TN's 72.48 calculations, used the 3-D and 2-D computational models Monte Carlo N-Particle (MCNP) and Discrete Ordinates Code (DORT).

Based on the information in NUH06L-0501, Rev. 1, and the 72.48 evaluation to which it is attached, TN did not provide a sufficient basis justifying that a change in the method used to estimate the accident dose rates in the calculation, supporting the 72.48 evaluation, was not a change in methodology. During the inspection, TN identified that a similar methodology had been used in section M.11.2.5.3 of the FSAR for the 32PT. However, this methodology was not properly referenced as a justification for the use of the methodology in the 72.48 evaluation. Additionally, TN had performed a calculation after completing the 72.48 evaluation, which showed that the simple extrapolation resulted in "essentially the same" result as the MCNP 3-D calculation. However, this calculation was not part of the 72.48 evaluation.

B. Weaknesses in Documentation of Evaluation Regarding TS 1.2.17a

As discussed in Section 3, the team identified a violation regarding TN's failure to obtain a CoC amendment for an issue that required a change to TS 1.2.11. In discussing this issue with TN, the team was provided the following information regarding TN's position on this issue. TN's main points were that:

1. it was not considered a change in method of evaluation, rather it was a change in an input parameter allowed to be made under 72.48,

2. other designs, like the 61BT and the 32PTH, have different initial temperatures of the basket and fuel cladding which provide examples of engineering judgement to determine an input parameter,

3. the 215°F initial temperature is still a valid initial temperature for the fuel cladding given the amount of time it would take to heat the fuel cladding from the 80-100°F range to the 215°F range (TN verbally stated that they had performed a back-of-the-envelope calculation showing an approximate cask heat-up rate of 4-10°F per hour),

4. LR 721004-321 Form 3.5-1 page 25 of 26 states no change to TS 1.2.17a is required because there are identical boundary conditions,

5. LR 721004-321 Form 3.5-3 page 18 of 22 evaluates departures from change in method of evaluation in the thermal area, and

6. there was no TS providing a time limit for bulk water removal.

As discussed in Section 3, the team determined that the FSAR, Section M.4.7.1, for the Standardized NUHOMS® design describes a methodology for determining the initial value for the temperature of the basket and fuel cladding at the start of vacuum drying. The discussion in Section M.4.7.1 is consistent with the sequence of operations described in Section M.8.1.3. This section of the FSAR discusses the DSC drying and backfilling process and that the inner top cover plate welding operation is done with water in the DSC. After the welding is completed and the dye penetrant weld examination is performed in accordance with TS 1.2.5, pumpdown of the cask is then performed.

In their evaluation for the OS197L TC, TN modified the procedure sequence in the FSAR, Section M.8.1.3, to allow for the water in the DSC to be pumped down much earlier in the process. The team's position is that this change in the procedural sequence for water pump down adversely affects the assumed initial cladding temperature used to calculate the TS allowed vacuum drying times. Specifically, TN's assumption in their evaluation that the 215° F cladding temperature is met at the beginning of vacuum drying is not conservative. The team considers that the initial fuel cladding temperature, at the start of vacuum drying in TN's revised pump down sequence, would be higher than the FSAR assumed value of 215° F because of the amount of time that the bulk of the water from the DSC is removed, when compared to that described in the FSAR. An assumed temperature of the fuel cladding higher than the 215°F basis in the FSAR would result in a shorter vacuum drying time than that specified in TS 1.2.17a. Therefore, the team's position is that TN should have sought a CoC amendment to reflect the shorter vacuum drying time limits that would result from the higher initial cladding temperature.

C. Weaknesses in Documentation of Evaluation Regarding TS 1.2.11

As discussed in Section 3, the team identified a violation regarding TN's failure to obtain a CoC amendment for an issue that required a change to TS 1.2.11. In discussing this issue with TN, the team was provided the following information regarding TN's position on this issue. TN's main points were that:

1. LR 721004-321 Form 3.5-1 page 19 through 21 provided an analysis supporting TN's conclusion that a change to TS 1.2.11 was not needed,

2. the TS is for ALARA only and not for detecting misloaded fuel, despite the stated objective of the TS,

3. the TS is redundant to fuel specification TS 1.2.1 and that if the proper fuel is loaded it will ensure that doses are ALARA,

4. TN tried to remove the TS in the past under Amendment 6 but was not successful,

5. NRC has agreed in the past that misload can be prevented via administrative controls through RAI 11-2 for the 32PT amendment 5 to the Standardized NUHOMS® design,

6. there was no direct analysis for the dose rates in the TS despite the SAR chapters referenced as the basis for the TS,

7. the objective of the TS improperly mentions fuel misloading and that this is a generic issue for multiple TSs, and

8. the SAR specifically allows the use of supplemental shielding in the axial direction and that use of supplemental shielding in the radial direction should also be allowed in meeting the TS.

The team considers that the scope of the changes made by TN under 72.48 goes against the scope of the changes envisioned by the 72.48 process. The statements of considerations (SOCs) associated with 10 CFR 72.48 (64 FR 53582) mention (ref. page 64 FR 53605) that: "This final rule allows licensees and certificate holders to make changes to the cask design, without obtaining prior NRC approval, for changes which do not significantly impact the ability of the cask to perform its intended function." The team considers that:

1. removing the lead shielding significantly impacted the ability of the OS197L TC to meet its design function of providing radiation shielding,

2. that the SOCs provide guidance for interpreting the regulations and they are referenced in both the rule and the regulatory guide associated with 10 CFR 72.48,

3. that the TC, that is considered part of the cask system, has two intended functions; a) allow for safe lifting and transfer of the DSC, and b) provide shielding,

4. that 10 CFR 72.3 provides a definition of a cask that is broad. Specifically, the definition states that "cask means all the components and systems associated with the container in which spent fuel or other radioactive materials associated with spent fuel are stored in an Independent Spent Fuel Storage Installation,"

5. the intended function of the cask system (i.e., provide shielding) is not met when the TC is used to transfer the spent fuel to the decontamination area and from the decontamination area to the transfer trailer, and

6. during remote operations, TN does not account for possible problems with the crane. Additionally, TN takes credit for shielding provided by buildings and structures that are outside the scope of the definition of the cask system.

D. Weaknesses in Documentation of Evaluation Regarding TS 1.2.1

As discussed in Section 3, the team identified a violation regarding TN's failure to obtain a CoC amendment for an issue that required a change to TS 1.2.1. In discussing this issue with TN, the team was provided the following information regarding TN's position on this issue. TN's main points were that:

1. as documented on page 17 of 26 of LR 721004-321, Rev 1, Form 3.5-1, this TS is not applicable to the transfer cask and the OS197L does not alter or further limit the fuel that can be stored,

2. the basis for the TS are not limits and per discussion with their legal counsel can be changed under 10 CFR 72.48 without the need for prior approval by the NRC,

3. the values specified in the basis for the 24P and 52B canisters are for a transfer cask with a solid neutron shield and these values could not be met using a transfer cask with a liquid neutron shield,

4. the TS would need to be changed to allow the draining of the transfer cask liquid neutron shield, which has been inspected by the NRC,

5. the staff previously inspected TN's 72.48 evaluation associated with the transfer cask with a liquid neutron shield and did not identify this TS basis statement as a problem,

6. the transfer cask dose rates stated in the basis of the TS are for ALARA purposes, and

7. despite the fact that the objective of the TS indicates that the specification, in part, serves to ensure that maximum surface doses are below the design limits, the purpose of the TS is for fuel qualification.

TN's 72.48 analysis (LR721004-321) states that the OS197L is designed to allow for the loading/unloading and transfer of the 24P, 52B, 61BT, 24PT2, 32PT and 24PHB. Therefore, the 24P and 52B are within the scope of TN's 72.48. Further, for this particular TS, the bases are written as part of the TS and are included in the attachment to the CoC. The bases of the TS explain that the radiological design criterion is that fuel stored in the NUHOMS system must not increase the average calculated TC surface dose rates beyond those calculated previously, and also states that the TC surface dose rates for the 24P and 52B should not exceed 598.1 mrem/hr. TN acknowledged that these dose rates would be exceeded if the OS197L TC was loaded with either the 24P or the 52B DSC. Therefore, the team concluded that a change to the TS was necessary to allow for the use of the OS197L TC.

6. Results of Other Issues Identified in Enclosure 3

Enclosure 3 was provided, by the staff, to TN prior to the inspection, and it served as a summary of the issues on which the team focused during the inspection. The purpose of this section of the inspection report is to document the staff's current disposition of the issues contained in Enclosure 3.

Resolved Issues

Issue 2 in Enclosure 3 involved TS 1.2.11, "Transfer Cask Dose Rates with a Loaded 24P, 52B, 61BT, or 32PT DSC." As discussed in Section 3, there is a violation associated with this TS. However, there was a second part of the issue, regarding when the surveillance measurement is to be taken. TN's 72.48 evaluation stated that the cask radial dose measurements are to be measured after the completion of vacuum drying, back filling, and sealing operations. The staff did not believe that this was consistent with ALARA principles because it could be several hours after the TC reaches the decontamination area before this measurement is taken. During discussions associated with this issue, TN referred the staff to several steps in the operating procedures where dose rates are taken to ensure ALARA precautions are taken. These steps were consistent with operating procedures already approved by the NRC in prior amendments. TN indicated that the official dose measurement to verify compliance with the TS in the dry condition has to be taken after the completion of vacuum drying, otherwise they could be challenged on how they determine that the cask is dry. Based on TN's arguments, the staff now considers this issue closed.

Issue 13 in Enclosure 3 involves spacers that were used in the DSC and the effect they would have on the axial dose rate measurements. TN indicated that the staff had previously determined (in Amendment 5, RAI 11-2) that, due to the use of administrative controls, misloading of fuel assemblies was not a safety concern. TN further described that the spacers were used at FC because FC's fuel was 20 inches shorter than the longest fuel that could be placed in the 32PTH DSC. The spacers were used for a hypothetical 10 CFR Part 71 application to prevent gross movement of the fuel. The spacers are made from thin walled stainless steel pipe and are located only at the top to the DSC. While the spacers would hinder identification of a misloaded assembly without recalculating the dose rate limits in TS 1.2.11, the staff agreed that because the issue was not safety significant (because fuel misloads are prevented through administrative controls), it did not warrant a change to the TS. Therefore, the staff considers this issue resolved.

<u>Issue Resolution</u>

Table 3 below provides a cross reference of the issues that were identified as open in Enclosure 3 at the beginning of the inspection and how these issues were resolved.

Table 3 - Resolution of Open Issues Contained In Enclosure 3

Item #	Description	Resolution
1	TS 1.2.17a "32PT DSC Vacuum Drying Duration Limit"	Violation Example 1
2	TS 1.2.11 "Transfer Cask Dose Rates with a Loaded 24P, 52B, 61BT, or 32PT DSC"	First part of issue: Violation Example 2; second part of issue: discussed in Section 6 of this Enclosure, no Violation
3	Thermal analysis associated with movement of the TC to decontamination area	Discussed in Section 4; also associated with Item #1 of this table
5	Thermal analysis associated with TC inside the transport trailer supplemental shielding	Discussed in Section 4 of this Enclosure
8	Simplified shielding analysis	Discussed in Section 5 of this Enclosure
12	TS 1.2.1 "Fuel Specification"	Violation Example 3
13	Affect DSC spacers have on axial dose measurements	Resolved without issuing a violation

<u>Documents Audited</u>

- TN LR No.: 721004-321, Revision 1, "10 CFR72.48 Applicability and 10 CFR 71 Review Form," dated March 31, 2006, including:
    - Form 3.5-1, Revision 0
    - Form 3.5-2, Revision 0

  - Form 3.5-3, Revision 0

- TN NUH06L-0400, Revision 1, “Thermal Analysis of OS197L and OS197L100 Transfer Casks,” dated March 23, 2006

- TN LR No.: 721004-338, Revision 1, “10 CFR 72.48 Applicability and 10 CFR 71 Review Form,” dated March 31, 2006

- TN NUH06L-0401, Revision 0, “CFD Evaluation of OS197L and OS197L100 Transfer Casks within Transfer Skid Auxiliary Shielding,” dated March 9, 2006

- TN NUH06L-0401, Revision 1, “CFD Evaluation of OS197L and OS197L100 Transfer Casks within Transfer Skid Auxiliary Shielding,” dated May 11, 2006

- TN NUH06L-0501, Revision 1, “OS197L 75 Ton Transfer Cask As-Built Configuration Shielding Analysis,” dated March 31, 2006

- TN NUH06L-0500, Revision 1, “Design of Integral Radiation Shield for On-Site Transfer Cask OS197-L and Calculation of Occupational Exposure due to 32PT DSC Design Basis Fuel,” dated March 14, 2006.

- TN interoffice Memo # 1121-06-082, “Documentation of information previously transmitted to OPPD for subsequent transmittal to NRC re: OPPD FCS Dry Fuel Storage,” dated August 30, 2006

- Certificate of Compliance No. 1004, including Attachment A, “Technical Specifications”

3. Inspection Procedures Used

IP 60857, “10 CFR 72.48, Evaluations”
NUREG/CR 6314, “Quality Assurance Inspections for Shipping and Storage Containers”

4. List of Acronyms Used

AHSM	Advanced Horizontal Storage Module
ALARA	As Low As Reasonably Achievable
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
DSC	Dry Shielded Canister
FC	Fort Calhoun Station
FSAR	Final Safety Analysis Report
HSM	Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OPPD	Omaha Public Power District
RG	Regulatory Guide
SAR	Safety Analysis Report

SER	Safety Evaluation Report
SFST	Division of Spent Fuel Storage and Transportation
TC	Transfer Cask
TN	Transnuclear, Inc.
TS	Technical Specifications

5. Persons Contacted

The team held an entrance meeting with TN personnel on August 28, 2006, to present the scope and objectives of the NRC inspection. On September 29, 2006, the team held an exit meeting with TN personnel to present the results of the inspection. The individuals present at the entrance and exit meetings are listed below in Table 1.

Table 1 Entrance and Exit Meetings Attendance

NAME	AFFILIATION	ENTRANCE	EXIT
James Pearson	NRC	X	X
Jorge Solis	NRC	X	X
Shana Helton	NRC	X	X
Joe Sebrosky	NRC	X	X
Bill Ruland	NRC	X	X
William Sutherland	TRANSNUCLEAR	X	X
Richard Flinn	TRANSNUCLEAR	X	
Steven White	TRANSNUCLEAR	X	
Tara Neider	TRANSNUCLEAR	X	X
Robert Grubb	TRANSNUCLEAR	X	X
Jayant Bondre	TRANSNUCLEAR	X	X
U. B. Chopra	TRANSNUCLEAR	X	X
Don Shaw	TRANSNUCLEAR	X	X
Prakash Narayanan	TRANSNUCLEAR	X	
James Axline	TRANSNUCLEAR	X	
Dan Kurtz	TRANSNUCLEAR	X	
Slava Guzeyev	TRANSNUCLEAR	X	
Greg Banken	CONSULTANT	X	

NOTICE OF VIOLATION

Transnuclear, Inc. Docket No. 072-1004
Columbia, MD 21045 CoC No. 1004

During an NRC inspection conducted on August 28 - September 29, 2006, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> 10 CFR 72.48(c)(1) states, in part, that a certificate holder may make changes in the facility or spent fuel storage cask design as described in the Final Safety Analysis Report (FSAR) (as updated) and make changes in the procedures as described in the FSAR (as updated) without obtaining ... (ii) a Certificate of Compliance (CoC) amendment submitted by the certificate holder pursuant to 10 CFR 72.244 if ... (B) a change in the terms, conditions, or specifications incorporated in the CoC is not required; and (C) the changes do not meet the criteria in 10 CFR 72.48(c)(2).
> Contrary to the above, in a 10 CFR 72.48 evaluation that was approved by Transnuclear Inc.(TN), a certificate holder, on March 31, 2006, TN failed to obtain a CoC amendment pursuant to 10 CFR 72.244 for changes made in the spent fuel storage cask design and changes in the procedures as described in the FSAR (as updated) and these design and procedure changes constituted a change in the terms, conditions, or specifications incorporated in the CoC. Specifically, although TN (1) eliminated lead shielding and otherwise modified the OS197L transfer cask design as described in the FSAR (as updated) to make it lighter; (2) provided additional temporary shielding in the decontamination area and on the transfer trailer to compensate for the reduction of the shielding; and (3) changed an operating procedure described in the FSAR (as updated) which allowed pump down of water from the dry shielded canister to occur much earlier in the process; TN failed to identify that the following technical specifications, which are incorporated in the CoC, would have required changes that needed prior NRC approval:
>
> 1. Technical Specification 1.2.17.a, “32PT DSC Vacuum Drying Duration Limit;”
>
> 2. Technical Specification 1.2.11, “Transfer Cask Dose Rates with a Loaded 24P, 52B, 61BT, or 32PT DSC;” and
>
> 3. Technical Specification 1.2.1, “Fuel Specification.”
>
> This is a Severity Level IV Violation (Supplement VI).

Pursuant to the provisions of 10 CFR 2.201, TN is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to Robert J. Lewis, Chief, Rules, Inspection and Operations Branch, Division of Spent Fuel Storage and Transportation, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a “Reply to a Notice of Violation” and should include for each violation: (1) the reason for the violation, or if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid further violations; and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's Agencywide Document Access and Management System (ADAMS) accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the basis for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post the Notice within 2 working days.

Dated this 9th day of November 2006.

72.48 Issues for Fort Calhoun Inspection
August 3, 2006

Note: Shaded area indicates issue is open

Revision 17

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
1	72.48(d)(1) inadequate documentation 72.48(c)(2)(viii)(departure from methodology described in the FSAR)	The staff believes that these criteria are tripped and a change to TS 1.2.17a for vacuum drying is needed. Section M.4.7.1 of the NUHOMS FSAR assumes vacuum drying initial conditions are 215 degrees F cladding temperature based on the DSC full of water. In actuality the canister will have been dry since it was drained in the spent fuel pool. The cask remains dry from the pumpdown in the pool through placement in the decontamination area and through the welding process. Fort Calhoun's contractor estimates 10 to 14 hours to complete this effort. Region IV's experience is that this effort is in the 24 hour range and that the Fort Calhoun contractor's estimate is optimistic. Regardless if the time is 14 hours or 24 hours, the initial fuel cladding temperature at the start of vacuum drying at Fort Calhoun would be considerably hotter than the FSAR assumed value of 215 degrees F. Therefore, the staff believes that the vacuum drying times in TS 1.2.17a should be recalculated.	Open

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
2	72.48(c)(1)(ii)(A) (technical specification change required)	The staff believes a technical specification (TS) change is required for Standardized NUHOMS® TS 1.2.11. TS 1.2.11 requires that the dose rate at 3 feet be ≤500 mrem/hour without water in the dry shielded canister (DSC), or ≤200 mrem/hr with water in the DSC. The values in this TS were calculated for the OS197 transfer cask (TC) in combination with various DSCs. Modifying the shielding in the TC requires that new cases be added to the TS to encompass the OS197L in combination with various DSCs. The basis for this technical specification limit is to maintain doses ALARA and to identify inadvertently loaded fuel assemblies. The dose rates in the TS were derived considering the design of the TC, including 3" of supplemental neutron shielding and 1" of steel on top of the TC during welding operations and no supplemental shielding in the radial direction. TN modified the OS197L transfer cask design to make the design lighter, including eliminating the lead shielding. To compensate for the reduction of the shielding in the transfer cask TN provides additional temporary shielding in the decontamination area. In LR 721004-321 form 3.5-3 section 5.1.3 TN provides a table of OS197L Normal Condition Dose Rates. The total dose rate for the OS197 surface dose rate is calculated to be 346 mrem/hour. In this same table the OS197L total surface dose rate with additional shielding is calculated to be 122 mrem/hour. TN contends that with the additional temporary shielding the OS197L provides better shielding. This statement is consistent with the results in the table in section 5.1.3 of the calculation. However, the TS value of 500mrem/hour is no longer conservative and a new lower value should be provided in this technical specification. Providing a new more conservative TS value in a license amendment is consistent with the guidance contained in NEI 96-07 appendix B. Regardless, the additional temporary shielding is not part of the TC design. The TS must be recalculated for the new design of the TC, without the temporary shielding. Additionally, the staff does not agree with the operating procedures suggested in LR No. 721004-321, Rev. 1, Form 3.5-1, Rev. 0, p. 6 of 26, which indicate that the cask radial dose measurements are to be measured after the completion of the vacuum drying, backfilling, and sealing operations. This is not consistent with surveillance requirements of technical specification 1.2.11, which states that the dose rate measurements are to be taken as soon as possible after the cask is removed from the spent fuel pool.	Open

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
3	72.48(d)(1) inadequate documentation 72.48(c)(2)(viii)(departure from methodology described in the FSAR)	The staff believes in the thermal area that these two criteria are tripped during the transfer of the OS197L TC from the spent fuel pool to the decontamination pad. The sequence of operations for the OS197L transfer cask that is described in LR 721004-321 is different from the sequence that is described and evaluated in the FSAR for the Standardized NUHOMS® design. Specifically, at Fort Calhoun TN proposed to drain the DSC completely before it leaves the spent fuel pool, and then move the DSC to the decontamination pad in this condition. In the FSAR for the Standardized NUHOMS® design up to 750 gallons of water (approximately ½ the water volume in the DSC) is allowed to be drained from the DSC prior to its removal from the spent fuel pool. In the FSAR there is step where this water is replaced in the canister after the transfer cask is in the decontamination area. TN contends that as long as the transfer cask annulus is maintained full of water they remain in an analyzed condition. During the transfer of the cask to the decontamination pad a pressurization tank of water is attached to the annulus as described in the FSAR. The purpose of this tank is to maintain a slightly positive pressure in the annulus so that spent fuel pool water does not come into contact with the outside surface of the DSC. This prevents contamination of the outside surface of the DSC. TN contends that this pressurization tank would serve as a makeup supply for any water that is lost due to boiling of the water in the annulus if the crane stops in the middle of the transfer from the spent fuel pool to the decontamination pad. TN estimates the water in the annulus to be about 1000 lbs. The weight of the water in the DSC if it were ½ full of water is approximately 3000 lbs. The additional water in the DSC in the FSAR provides additional margin for protection of the fuel cladding if the crane were to stop during the transfer from the spent fuel pool to the decontamination pad. The staff believes that completely draining the water in the DSC is non-conservative relative to what was assumed and described in the FSAR and trips criteria vii and viii of 10 CFR 72.48(c)(2).	Open

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
4	72.48(c)(2)(vii) (design basis limit for fission product barrier) and 72.48(c)(2)(viii)(departure from methodology described in the FSAR)	The staff is still evaluating this issue. The staff believes that these two criteria maybe tripped when the OS197L TC is moved from the decontamination pad to the transport trailer. The transfer cask has the neutron shield drained during this evolution and the annulus full of water. TN has performed a preliminary calculation that shows the annulus could boil off in 8 to 10 hours. In TN's response they state that water will be maintained in the transfer cask annulus and that by doing this the transient (movement of the TC from the decontamination area to the trailer) is bounded by the UFSAR analysis. The staff agrees with TN's response, therefore, this issue is closed.	Closed

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
5	72.48(c)(2)(viii)(departure from methodology described in the FSAR)	The staff believes that there is a change in the thermal methodology that is introduced as a result of the introduction of additional shielding on the transfer trailer. TN states in LR 721004-321 Form 3.5-3 Revision 0 page 18 of 22 that the "analysis of the OS197L transfer cask within the shielded transfer skids used a similar methodology as the analysis of the COC 1029 , Amendment 1 AHSM." TN goes onto state that "although the mesh arrangement used for each analysis was, by necessity, different to reflect the differences in geometry, similar types of mesh elements and aspect ratio are used for both analyses." The staff notes that the CoC 1029 Amendment 1 does not model a transfer cask inside an AHSM. The staff believes that the addition of the transfer cask inside the AHSM is a change in methodology and results in non-conservative results. (TN did mention verbally that in some cases that transfer trailer shielding would provide some reduction in temperature from solar heating of the DSC). The staff further believes that in addition to changes in the mesh arrangements TN had to change its thermal model to account for the transfer cask inside the transfer trailer shielding. Taken together the staff believes this constitutes a change in methodology which trip criterion viii of 10 CFR 72.48(c)(2). (See supplemental note that further describes this basic problem). TN supplemented their response in a white paper on 4/12/06. Staff also reviewed calculations on 4/19/06. The staff disagrees with TN's assessment that a methodology change was not made. TN states the following regarding its licensing basis: The Fluent code CFD methodology is used to calculate the OS197L transfer cask shell temperature while the transfer cask is on the transfer trailer in a horizontal position with the trailer/skid shielding in place. These shell temperatures are then used to estimate the fuel cladding temperature in a separate calculation. As described in the TN's 72.48 Evaluation the CFD methodology was used to determine the DSC shell temperature distribution when the DSC is inside the Advanced NUHOMS® HSM (AHSM). The CFD methodology used for the analysis of this configuration is not a departure from this NRC approved method of evaluation. The OS197L analysis represents a change in geometry (Continued on next page)	Open

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
5 Continued	see previous page	(continued from previous page) and not a change in method. The similarity in the two geometries analyzed is documented in the Question 8 Response in LR 721004-321, Revision 1, Form 3.5-3 discussion for calculation NUH06L-401. As discussed in Reference [2], the approved CFD methodology was used with appropriate constraints and limits. Mesh sensitivity analysis was performed to assure that the results are mesh independent. Inputs to the CFD modeling (geometry) are different due to differences in dimensions, however, the modeling methods are the same. NRC disagrees with TN's assessment. As documented in Calculation No. SCE-23.410 (Thermal Analysis of NUHOMS® Advanced Horizontal Storage Module Using CFD Method), the Advanced NUHOMS® HSM (AHSM) CFD analysis is based on Fluent™/Icepak™. On the other hand, the analysis provided on Calculation No. NUH06L-401 (CFD Evaluation of OS197L 100 Transfer Casks within Transfer Skid Auxiliary Shielding) is based on Fluent™ and Gambit™ Codes. This constitutes a change on the computational methods. (Continued on next page) Also, TN used a different CFD model for the transfer cask inside the skid shielding. Additionally, per Section 9.1.3 of Calculation No. NUH06L-0400 (Thermal Analysis of OS197L and OS197L100 Transfer Casks), the temperatures for the different components of the transfer cask and DSC, including the maximum fuel cladding temperature , were obtained by extrapolation of data (which is basically a hand calculation). This approach differs significantly from the FSAR approach. Therefore, based on the above, the staff believes TN's use of the ASHM methodology to model the transfer cask inside the supplemental shielding while on the transfer trailer constitutes a departure from the FSAR methodology which trips criterion viii of 10 CFR 72.48(c)(2).	

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
6	72.48(c)(2(v) introduction of possibility for an accident of a different type 72.48(c)(2)(vi) possibility of a malfunction of an SSC with a different result	The staff believes this issue is resolved and that no 72.48 criteria are tripped. Below is background describing the issue and the basis for its closure. The staff believes that the remote operation of the crane (due to high radiation doses) trips this criteria. The staff disagrees with TN's assessment contained on page 17 of LR 721004-321 Form 3.5-3. The remote operations that TN describes are of movement of the transfer cask from the spent fuel pool to the decontamination pad. Placement of the transfer cask in a shield sleeve in the decontamination pad. Disconnecting the crane from the transfer cask in the decontamination pad, picking up and then placing a shield bell over the top portion of the transfer cask. After the DSC is welded there is additional remote removal of the shield bell. Remote pickup and movement of the transfer cask from the decontamination pad and downending the transfer cask onto the transfer trailer. None of this remote operation is currently described in the FSAR. Malfunction of the SSCs that are possible because of the remote operation include misalignment of the transfer cask in the decontamination pad, misalignment on the transfer trailer, and sticking of the crane during the downending operation. The staff does not believe that these possibilities have been adequately addressed and that the criteria vi of 10 CFR 72.48(c)(2) is tripped because of the introduction of the remote operation of the crane. The staff also believes that the introduction of cameras and lasers introduces or creates the possibility for a malfunction of an SSC important to safety. In this case if the lasers or cameras are misaligned it could cause the misplacement of the transfer cask in the decontamination pad or misalignment on the transfer trailer. TN responded to the above by stating that this type of equipment is routinely used during placement of the transfer cask in the spent fuel pool, and therefore, it does not introduce the possibility of a new accident. The staff accepts this argument.	Closed

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
7	72.48(c)(1)(ii)(B) (Certificate of Compliance change required)	The staff is closing this issue and subsuming the remaining parts into issue 2. The staff believes that this criteria is tripped because the Certificate of Compliance (CoC) for the Standardized NUHOMS FSAR provides a description of the transfer cask. The staff believes that the definition of the transfer cask needs to be expanded to include the temporary shielding (i.e., lower 6 inch steel sleeve and shield bell). The basis for the staff believing that this shielding needs to be provided in the description of the transfer cask in the CoC is that TN takes credit for this shielding when it verifies compliance with technical specification 1.2.11 The portion of the issue dealing with definition of the transfer cask is subsumed into issue number 2.	Closed
8	72.48(c)(2)(viii)(departure from methodology described in the FSAR)	During review of shielding calculation for the OS197 transfer cask (NUH06L-0501, Rev. 1) the staff identified a change in methodology in the shielding calculation. Chapter M.5.4 of the Standardized NUHOMS SAR describes the shielding calculations that were performed for the 32PT DSC and its associated transfer cask. This shielding evaluation was done for loading, transfer and storage of the 32PT DSC for normal, off-normal, and accident conditions. The methodology described in Chapter M.5.4 of the SAR used computational models MCNP and DORT. The methodology is described in section M.5.4.5 of the SAR for the standardized NUHOMS SAR. In calculation NUHO6L-0501, Rev. 1 the analysis for the accident condition involving the loss of the inner skin of the neutron shield used a simple extrapolation using a ratio of dose rates derived from prior MCNP calculations. This simple calculation does not appear to be conservative. That is, the extrapolation is a one-dimensional calculation (the MCNP and DORT models were 3-D and 2-D), which does not account for changing energy-dependence as radiation passes through the shielding. This represents the use of a different mathematical model not previously approved by the NRC. While the change of methodology was only used for analyzing a small portion of the accident conditions, changing the methodology trips the criteria of 72.48(c)(2)(viii).	Open

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
9	72.48(c)(2)(viii)(departure from methodology described in the FSAR) 72.48(c)(1)(ii)(A) (technical specification change required)	The staff believes that there is a change in methodology based on TN's assumption that the design basis (i.e., bounding) fuel for the OS197L transfer cask configuration is the same as that for the OS197. Changing the design basis fuel in the shielding analysis is a change of methodology as defined in NEI 72.48 guidance. The staff agrees that the design basis fuel for the 1.2 kW assemblies is unchanged. However, the design basis fuel for the 0.6 kW assemblies has changed, as documented on p. 107 and 125 of 147 in TN calculation NUH06L-0500. The staff does not agree with TN's assertion that this can be ignored. SAR section M.5 states that the bounding (design-basis) burnup, minimum initial enrichment and cooling time combinations used in the shielding evaluations for the 0.6kW fuel assemblies are: - 45 GWd/MTU, 3.3 wt.% U-235, 23-year cooled - for the inner sixteen assemblies in the TC models. TN states on page 107 of its NUH06L-0500 calculation that the design basis inner sixteen fuel is bounding, except for configuration B. Configuration B is the OS-197L with no supplemental shielding. The bounding fuel for configuration B for the inner 16 assemblies as shown on page 125 of TN NUH06L-0500 is: - 25GWd/MTU, 2.1 wt. % U-235, 6 year cooled The calculated dose value for the design basis fuel is 103.1 mrem/hr while for the limiting fuel for configuration B the calculated dose value is 314.6 mrem/hr (these values are shown on page 125 of NUH06L-0500. The staff believes that changing the design basis fuel for the shielding analysis requires prior NRC approval. TN contends that the total dose rate on the side of the transfer cask is driven by the outer 16 fuel assemblies and that inner 16 fuel assemblies contribute only a small fraction of the total dose and that prior NRC approval is not needed for the change in design basis fuel for the inner 16 fuel assemblies.	Closed

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
10	72.48(c)(1)(ii)(B) (Certificate of Compliance change required)	The staff believes this issue is closed. Below is background describing the issue and the basis for its closure. Related to item 7 above. Certificate of Compliance condition 3.b states in part that "the standardized NUHOMS® System is certified as described in the final safety analysis report (FSAR) and in the NRC's Safety Evaluation Report (SER). The transfer cask as described in the FSAR and the NRC's Safety Evaluation report looks nothing like the OS197L transfer cask described in TN's 72.48. Major changes made to the OS197L transfer cask include: the elimination of the TC lead shielding, TC liquid neutron shield made in two sections, and provisions and reliance on supplemental shielding in the decontamination area and on the trailer/skid. The reliance on the supplemental shielding goes beyond temporary shielding. The 10 CFR Part 20 occupational dose limits could not be met without this supplemental shielding.	Closed
11	72.48(c)(2(v) introduction of possibility for an accident of a different type 72.48(c)(2)(vi) possibility of a malfunction of an SSC with a different result	The staff believes this issue is closed. Below is background describing the issue and the basis for its closure. One of the changes made to the OS197L transfer cask is that the neutron shield is made in two sections instead of one section. The staff can not determine based on TN's 72.48 if the difference in the construction of the neutron shield introduces the possibility of a new accident. For example how does a neutron shield that is made in two sections instead of one respond structurally to a drop. Is there a possibility that the neutron shield can become separated and become a missile, etc.. While the design concept of two wrap-around exterior independent neutron shield tank halves is a departure from a previous design, the design criteria are that the accidental transfer cask drop is considered with the resulting loss of the neutron shield. The previous structural shell was 1.5 inches thick with the neutron shield tank separate steel outer shell thickness of 3/16". The OS197L is to have a structural shell 2.68 inches thick with the inner and outer steel shells of the neutron shield tank having thicknesses of 1/4". The staff's judgment is that the new neutron shield tank does not pose a structural threat to the transfer cask as a missile or in any other mode of loading resulting from an accidental cask drop.	Closed

Item #	10 CFR 72.48 Criteria	Background/Basis for non compliance	Status
12	72.48(c)(1)(ii)(A) (technical specification change required)	The staff believes a TS change is required for TS 1.2.1 Tech spec 1.2.1, "Fuel Specification" contains the following under its basis: "The radiological design criterion is that fuel stored in the NUHOMS® system must not increase the average calculated HSM or transfer cask surface dose rates beyond those calculated for the 24P, 24PHB, 52B, 61BT, or 32PT canister full of design basis fuel assemblies with or without BPRAs. The design value average HSM and cask surface dose rates for the 24P and 52B canisters were calculated to be 48.6 mrem/hr and 591.8 mrem/hr respectively based on....." Page 1 of 26 of TN's LR721004-321 states that the design intent of the OS197L is to allow for the loading/unloading and transfer of the licensed DSCs (24P, 52B, 61BT, 24PT2, 32PT, and 24PHB) and maintain the bounding crane load to less than 75 tons. Therefore the staff believes that the 24P and 52B canisters are within the scope of TN's 72.48. The dose rate values provided in TS 1.2.1 for the 24P and 52B of 48.6 mrem/hr and 591.8 mrem/hr are two orders of magnitude lower than the dose rate values would be for a OS197L transfer cask based on the 32PT analysis. Therefore, the values in TS 1.2.1 are no longer applicable for the OS197L transfer cask and a TS change is required. The staff also disagrees with TN's response to this item that these dose rate values are applicable to the storage condition (HSM) and long duration transfer conditions. The staff believes that this explanation goes against the clear language of this TS which references the TC surface dose rate. In the staff's opinion the OS197L TC dose rate clearly exceeds those calculated for the OS197 transfer cask and a TS change is needed.	Open
13	72.48(c)(1)(ii)(A) (technical specification change required)	The staff understands that at Fort Calhoun spacers were used in the dry shielded canister because the Fort Calhoun fuel assemblies are short. The staff does not understand how the spacer design was evaluated in the dose calculations that support TS 1.2.11 values.	Open

NOTE: The following information is an updated excerpt from NRC Information Notice 96-28 issued in 1996.

NRC INFORMATION NOTICE 96-28

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
WASHINGTON, D.C. 20555

May 1, 1996

NRC INFORMATION NOTICE 96-28: SUGGESTED GUIDANCE RELATING TO DEVELOPMENT AND IMPLEMENTATION OF CORRECTIVE ACTION

Addressees

All material and fuel cycle licensees.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to provide addressees with guidance relating to development and implementation of corrective actions that should be considered after identification of violation(s) of NRC requirements. It is expected that recipients will review this information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not new NRC requirements; therefore, no specific action nor written response is required.

Background

On June 30, 1995, NRC revised its Enforcement Policy, to clarify the enforcement program's focus by, in part, emphasizing the importance of identifying problems before events occur, and of taking prompt, comprehensive corrective action when problems are identified. Consistent with the revised Enforcement Policy, NRC encourages and expects identification and prompt, comprehensive correction of violations.

In many cases, licensees who identify and promptly correct non-recurring Severity Level IV violations, without NRC involvement, will not be subject to formal enforcement action. Such violations will be characterized as "non-cited" violations as provided in Section VI.A of the Enforcement Policy. Minor violations are not subject to formal enforcement action. Nevertheless, the root cause(s) of minor violations must be identified and appropriate corrective action must be taken to prevent recurrence.

If violations of more than a minor concern are identified by the NRC during an inspection, licensees will be subject to a Notice of Violation and may need to provide a written response, as required by 10 CFR 2.201, addressing the causes of the violations and corrective actions taken to prevent recurrence.

In some cases, such violations are documented on Form 591 (for materials licensees) which constitutes a notice of violation that requires corrective action but does not require a written response. If a significant violation is involved, a predecisional enforcement conference may be held to discuss those actions.

The quality of a licensee's root cause analysis and plans for corrective actions may affect the NRC's decision regarding both the need to hold a predecisional enforcement conference with the licensee and the level of sanction proposed or imposed.

<u>Discussion</u>

Comprehensive corrective action is required for all violations. In most cases, NRC does not propose imposition of a civil penalty where the licensee promptly identifies and comprehensively corrects violations. However, a Severity Level III violation will almost always result in a civil penalty if a licensee does not take prompt and comprehensive corrective actions to address the violation.

It is important for licensees, upon identification of a violation, to take the necessary corrective action to address the noncompliant condition and to prevent recurrence of the violation and the occurrence of similar violations. Prompt comprehensive action to improve safety is not only in the public interest, but is also in the interest of licensees and their employees. In addition, it will lessen the likelihood of receiving a civil penalty. Comprehensive corrective action cannot be developed without a full understanding of the root causes of the violation.

Therefore, to assist licensees, the NRC staff has prepared the following guidance, that may be used for developing and implementing corrective action. Corrective action should be appropriately comprehensive to not only prevent recurrence of the violation at issue, but also to prevent occurrence of similar violations. The guidance should help in focusing corrective actions broadly to the general area of concern rather than narrowly to the specific violations. The actions that need to be taken are dependent on the facts and circumstances of the particular case.

The corrective action process should involve the following three steps:

1. <u>Conduct a complete and thorough review of the circumstances that led to the violation.</u> Typically, such reviews include:

    - Interviews with individuals who are either directly or indirectly involved in the violation, including management personnel and those responsible for training or procedure development/guidance. Particular attention should be paid to lines of communication between supervisors and workers.

    - Tours and observations of the area where the violation occurred, particularly when those reviewing the incident do not have day-to-day contact with the operation under review. During the tour, individuals should look for items that may have contributed to the violation as well as those items that may result in future violations. Reenactments (without use of radiation sources, if they were involved in the original incident) may be warranted to better understand what actually occurred.

    - Review of programs, procedures, audits, and records that relate directly or indirectly to the violation. The program should be reviewed to ensure that its overall objectives and requirements are clearly stated and implemented. Procedures should be reviewed to determine whether they are complete, logical, understandable, and meet their objectives (i.e., they should ensure compliance with the **current** requirements). Records should be reviewed to determine whether there is sufficient documentation of necessary tasks to provide an record that can be audited and to determine whether similar violations have occurred previously. Particular attention should be paid to training and qualification records of individuals involved with the violation.

2. Identify the root cause of the violation.

   Corrective action is not comprehensive unless it addresses the root cause(s) of the violation. It is essential, therefore, that the root cause(s) of a violation be identified so that appropriate action can be taken to prevent further noncompliance in this area, as well as other potentially affected areas. Violations typically have direct and indirect cause(s). As each cause is identified, ask what other factors could have contributed to the cause. When it is no longer possible to identify other contributing factors, the root causes probably have been identified. For example, the direct cause of a violation may be a failure to follow procedures; the indirect causes may be inadequate training, lack of attention to detail, and inadequate time to carry out an activity. These factors may have been caused by a lack of staff resources that, in turn, are indicative of lack of management support. Each of these factors must be addressed before corrective action is considered to be comprehensive.

3. Take prompt and comprehensive corrective action that will address the immediate concerns **and** prevent recurrence of the violation.

   It is important to take immediate corrective action to address the specific findings of the violation. For example, if the violation was issued because radioactive material was found in an unrestricted area, **immediate** corrective action must be taken to place the material under licensee control in authorized locations. After the immediate safety concerns have been addressed, timely action must be taken to prevent future recurrence of the violation. Corrective action is sufficiently comprehensive when corrective action is broad enough to reasonably prevent recurrence of the specific violation as well as prevent similar violations.

In evaluating the root causes of a violation and developing effective corrective action, consider the following:

1. Has management been informed of the violation(s)?

2. Have the programmatic implications of the cited violation(s) and the potential presence of similar weaknesses in other program areas been considered in formulating corrective actions so that both areas are adequately addressed?

3. Have precursor events been considered and factored into the corrective actions?

4. In the event of loss of radioactive material, should security of radioactive material be enhanced?

5. Has your staff been adequately trained on the applicable requirements?

6. Should personnel be re-tested to determine whether re-training should be emphasized for a given area? Is testing adequate to ensure understanding of requirements and procedures?

7. Has your staff been notified of the violation and of the applicable corrective action?

8. Are audits sufficiently detailed and frequently performed? Should the frequency of periodic audits be increased?

9. Is there a need for retaining an independent technical consultant to audit the area of concern or revise your procedures?

10. Are the procedures consistent with current NRC requirements, should they be clarified, or should new procedures be developed?

11. Is a system in place for keeping abreast of new or modified NRC requirements?

12. Does your staff appreciate the need to consider safety in approaching daily assignments?

13. Are resources adequate to perform, and maintain control over, the licensed activities? Has the radiation safety officer been provided sufficient time and resources to perform his or her oversight duties?

14. Have work hours affected the employees' ability to safely perform the job?

15. Should organizational changes be made (e.g., changing the reporting relationship of the radiation safety officer to provide increased independence)?

16. Are management and the radiation safety officer adequately involved in oversight and implementation of the licensed activities? Do supervisors adequately observe new employees and difficult, unique, or new operations?

17. Has management established a work environment that encourages employees to raise safety and compliance concerns?

18. Has management placed a premium on production over compliance and safety? Does management demonstrate a commitment to compliance and safety?

19. Has management communicated its expectations for safety and compliance?

20. Is there a published discipline policy for safety violations, and are employees aware of it? Is it being followed?

This information notice requires no specific action nor written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below.

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