PROPOSED STANDARD TECHNICAL SPECIFICATIONS (STS) FOR FUEL SPECIFICATIONS

2.1.1 Fuel to be Stored in the [CASK] System

2.1.1 For each loading of fuel in the [CASK] system, K_{eff} shall be α 0.95 for all normal, off-normal and hypothetical accident conditions, including all biases and uncertainties.

Basis:

Key fuel parameters are identified in the SAR for the [ ] DCSS, as applicable, for each fuel type analyzed to be stored. Applicability is determined for the particular fuel type (for example, very few fuel types would employ partial length fuel rods in their design). These are either explicitly identified for each fuel assembly design or are representative of a class of fuel assemblies that are grouped together for analytical purposes to define a fuel type (e.g., Westinghouse 17x17, Siemens 17x17, B&W 17x17 that was used in a particular reactor or reactor type).

The design criteria, assumptions and conservatisms utilized in the development of the analytical models for the criticality safety analyses are explicitly defined and will be maintained in the SAR for the [ ] DCSS.

Sensitivity or case studies have been performed to determine bounding values of parameters or modeling assumptions to be used in criticality safety analyses and are defined and will be maintained in the SAR for the [ ] DCSS in sufficient detail to document and detail the effect of the analyzed variations on K_{eff}. The bounding values or assumptions derived from the studies are identified. Studies identify the fuel assembly or canister arrays which are used, as appropriate, to define the bounding effects.

The methodology employed for performing criticality safety analyses has been found acceptable by the NRC, and is defined in sufficient detail and will be maintained in the SAR for the [ ] DCSS.

GUIDELINES FOR THE IMPLEMENTATION OF FUEL SPECIFICATIONS STANDARD TECHNICAL SPECIFICATIONS (STS)

APPENDIX [ ] TO NEI 99-06

In order for the Fuel Specifications Standard Technical Specifications (STS) to be employed for a given Spent Fuel Management System (DCSS), the guidelines contained in this appendix define the information that must be contained and maintained in the Safety Analysis Report (SAR) in order to ensure that any changes to analytical methods for criticality calculations or to the fuel types that are to be stored in the DCSS under 10 CFR 72.48 are bounded in such a way so that the subcriticality limit of 0.95 Keff is not exceeded.

Existing SARs, for DCSS that have been certified by the NRC, may not all meet the guidelines for criticality safety determinations established in this document. The certificate holder may choose to either upgrade the SAR to be consistent with the guidelines in order to employ the STS, or maintain the current certificate and its custom Technical Specifications for fuel specifications. The SAR upgrade would likely be subject to NRC review and approval. Applications for new DCSS would employ the guidelines in full.

These guidelines have been developed based on discussions with the NRC and through review of DCSS certification documents for the latest generation of DCSS from four vendors; NAC, TN, Holtec, and FuelSolutions. It is expected that these guidelines will give the NRC sufficient assurance that all parameters important to the maintenance of criticality safety will be rigorously controlled and that changes to the DCSS, fuel parameters or the types of fuel stored in the DCSS will be conservatively controlled under the requirements of 10 CFR 72.48, and that NRC review and approval of changes that may adversely impact criticality safety will be assured.

1. Fuel Parameters

   The following fuel parameters will be identified in the SAR, as applicable, for each fuel type analyzed to be stored. Applicability will be determined for the particular fuel type (for example, very few fuel types would employ partial length fuel rods in their design). These may be explicitly identified for each fuel assembly design or may be representative of a class of fuel assemblies that are grouped together for analytical purposes to define a fuel type (e.g., Westinghouse 17x17, Siemens 17x17, B&W 17x17 that was used in a particular reactor or reactor type).

   a. clad material
   b. initial enrichment (i.e., maximum pin enrichment, bundle average enrichment, lattice average enrichment, etc.)
   c. pellet or stack UO_2 density

d. number of fuel rods, including number of partial length rods
e. clad O.D.
f. clad thickness
g. pellet diameter
h. fuel rod pitch
i. active fuel length
j. number and location of water holes (water rods)
k. number of inert (solid) rods
l. distance from bottom of the fuel assembly to start of active fuel
m. number, size, material, and location of guide and instrument tubes
n. fuel channels – material, presence and thickness
o. maximum uranium loading (total)
p. presence of burnable poison and control rod assemblies

Limitations on the type of fuel assemblies (Intact, Mixed Oxide, Partial and Damaged) that have been analyzed for storage will be identified as it pertains to its effect on cask criticality analyses.

2. Analysis and Model Design Criteria, Assumptions and Conservatisms

The design criteria, assumptions and conservatisms utilized in the development of the analytical models for the criticality safety analyses will be explicitly defined and controlled in the SAR. An example[1] of these criteria, assumptions and conservatisms are:

- The canisters are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.

- No credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product poisons.

- The criticality analyses assume [75%] of the manufacturer's minimum Boron-10 content for the borated neutron absorber.

- The fuel stack density is conservatively assumed to be [96%] of theoretical ($10.522\ g/cm^3$) for all criticality analyses.

- No credit is taken for the ^{234}U and ^{236}U in the fuel.

- When flooded, the moderator is assumed to be pure, unborated water at a

[1] This example is a composite of design criteria, assumptions and conservatisms taken from several certification documents and is not meant to imply a minimum or required list for any cask vendor, but rather is provided for illustrative purposes to show the detail that is typically addressed in current DCSS criticality safety analyses.

temperature corresponding to the highest reactivity within the expected operating range (i.e., water density of 1.000 g/cc).

- When flooded with borated water (for certain DCSS designs), the optimum borated water density will be determined.

- Neutron absorption in minor structural members and heat conduction elements is neglected, i.e., spacer grids, basket supports, and heat conduction elements are replaced by water (if this is demonstrated to be conservative for the specific design).

- Evaluation of the reactivity impact for a variety of channel dimensions in the BWR most-reactive-assembly analysis to demonstrate the impact of the channel material on cask criticality.

- In compliance with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values) is assumed.

- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded.

- Planar-averaged enrichments are assumed for BWR fuel. (In accordance with NUREG-1536, analysis is presented to demonstrate that the use of planar-average enrichments produces conservative results.)

- In accordance with NUREG-1536, fuel-related burnable neutron absorbers, such as the Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.

- For evaluation of the bias, all benchmark calculations that result in a k_{eff} greater than 1.0 are conservatively truncated to 1.0000, in accordance with NUREG-1536.

- For fuel assemblies that contain low-enriched axial blankets, the governing enrichment is that of the highest planar average, and the blankets are not included in determining the average enrichment.

- For intact fuel assemblies, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

- Full and partial loading configurations for the canister are analyzed.

- The radial boundary can be defined as either the transport cask body outer shell (normal conditions) or the transfer cask shell with the appropriate neutron shielding for normal and accident conditions. The single package model is surrounded by [ ] inches of water for reflection. The multiple package array model consists of an infinite number of canisters/casks in a close packed arrangement (triangular pitch array) with the adjacent casks in contact with one another.

- Modeling of the canister axially from the top of the bottom end inner closure plate to a point just below the top shield plug support ring. Reflected planes are inserted at these points to prohibit neutron leakage thus maximizing K_{eff}.

- Both normal conditions and hypothetical accident conditions are evaluated. The normal condition models of the DCSS include consideration of: a) complete flooding with water at a density sufficient for optimum moderation; b) worst case asymmetric assembly placement within the guide/fuel tubes; and c) application of worst case material and fabrication tolerances. The hypothetical accident condition models include all the normal conditions as well as the addition of a permanent deformation of guide/fuel tubes between support plates, the axial detachment of the guide/fuel tubes from the basket structure, and the loss of the transportation cask neutron shield assembly.

3. Studies will be performed to determine bounding values of parameters or modeling assumptions to be used in criticality safety analyses and will be presented and maintained in the SAR in sufficient detail to document and detail the effect of the analyzed variations on K_{eff}. The bounding value or assumption derived from the studies will be identified, as well as the fuel assembly or canister arrays which are used, as appropriate, to define the bounding effects. Examples of the parameters and modeling assumptions that are typically subjected to these studies are:

- Enrichment – lattice, pin (BWR), pellet (damaged fuel can analyses)
- Clad OD
- Clad thickness (or clad ID)
- Pellet diameter
- Fuel rod pitch
- Active fuel length
- Fuel channel thickness
- Borated water draindown (if soluble boron concentration is required in loading)
- Bounding configuration (storage, transfer, transportation)
- Single vs. multiple cask package array analyses to determine most reactive configuration
- Limiting canister design
- Preferential flooding/draining of fuel assemblies in overall canister vs. fuel assemblies stored in damaged fuel cans
- Interspersed and interior moderator density analyses to determine optimum moderator density

- Effects of loading one or more higher enrichment fuel assemblies than is allowed for the canister maximum enrichment
- Treatment of special fuel types, such as mixed-oxide, partial (fuel rods missing from the lattice), or damaged fuel assemblies, including mixed loading analyses for the canister
- Treatment of partial-length fuel rods and axially-blanketed fuel rods

These studies are present, as necessary, to support the following process that is typically followed in demonstrating the criticality safety of the cask contents:

- Evaluation of each of the proposed contents to determine the most reactive (bounding) fuel, to be used in all subsequent analyses;
- Evaluation of the most reactive configuration of the fuel and basket, with variables considered such as location of fuel in the compartment, the dimensions of the basket components, and the presence of moderator;
- Evaluation of special contents, such as damaged or partial fuel assemblies.

4. The methodology employed for performing criticality safety analyses will be defined in sufficient detail and maintained in the SAR. This includes the following:

   - Use of an accepted calculational methodology, such as MCNP-4a, SCALE 4.3 CSAS, or other. This will include definition of any cross-section libraries and other features that must be controlled to ensure that the calculational methodology will be consistently maintained over time. Additionally, the calculational platform (computer system) must be maintained as part of the certificate holder's software QA program. Any deviations or changes to any aspect of the calculational methodology must be analyzed in accordance with 10 CFR 72.48 to determine if a methodology change requires NRC review and approval, if applied to criticality safety analyses performed subsequent to those approved for the certified DCSS.
   - Description of the analyses performed to benchmark the calculational methodology to critical experiments to arrive at the bias to be applied to K_{eff} analyses.
   - Definition and description of the radial and axial criticality models used for analyzing the normal and hypothetical accident conditions.
   - Description of how the limiting Upper Subcritical Limit (or an alternative approach) for a range of parameter values is determined, based on consideration of the following parameters, as appropriate to the DCSS design and the fuel to be stored:

     a. assembly pin pitch
     b. enrichment
     c. water-to-fuel volume ratio
     d. H-to-^{235}U ratio

e. B-10 concentration in any neutron poison
f. Percentage of fissile material that is Pu (as opposed to U) for MOX fuel storage

- Demonstration of source convergence considering source location and number of histories.
- Description of differences between the calculational model and the physical design of the DCSS.

CONTROL OF CHANGES TO FUEL SPECIFICATIONS UNDER 10 CFR 72.48

If the level of detail presented in the guidelines is contained in the Safety Analysis Report, then changes to the design of a DCSS or to the fuel contained in the DCSS can be controlled under the requirements of 10 CFR 72.48. Some examples follow that show how changes could fall within the authority of the DCSS vendor to implement without prior NRC review and approval, as well as examples of changes that would require prior NRC review and approval prior to implementation.

Examples of Changes That Would Not Require Prior NRC Review and Approval

- The DCSS vendor is supplying a DCSS to utility XYZ for storage of 17x17 fuel utilized in a Westinghouse reactor. The utility has SNF that it wishes to store that has a fuel design from a different vendor that is compatible with fuel assembly designs that are currently enveloped in the SAR criticality safety analyses, but has an improved cladding material and improved fuel grids. The computational model used by the DCSS vendor in performing criticality safety analyses is used to perform bounding studies for the new fuel design. These analyses demonstrate that the new fuel design is bounded in all aspects by an existing analyzed fuel type approved for storage in the DCSS. The 10 CFR 72.48 analysis of this change would show that it may be implemented without prior NRC review and approval, since no methodology changes resulted, subcriticality of the DCSS with the new design was maintained, and no Technical Specification change was required.
- A DCSS vendor has selected a different fabricator for the borated neutron absorbers used in the fuel canister design. The manufacturing and testing processes that the fabricator uses to control product specifications and verify minimum boron content are improved to where the criticality analyses can assume 90% of the minimum B^{10} content as opposed to 75% as assumed in the NRC reviewed and certified DCSS. The DCSS vendor wants to revise the borated neutron absorber specifications to reduce the amount of material required for each absorber plate that would be installed in the canister as a result of this improvement. The criticality analyses performed by the vendor show that the revised specifications do not cause the conclusions of the analyses to change. The 10 CFR 72.48 analysis of this change would show that it may be implemented without prior NRC review and approval, since no methodology changes resulted, subcriticality of the DCSS with the new design was maintained, and no Technical Specification change was required. [NOTE: This example was predicated on NRC's planned issue of an ISG that allows 90% instead of 75% minimum as specified in NUREG-1536]
- During review of the SNF assembly configurations contained in the spent fuel pool that are planned to be loaded into a certified DCSS, it is discovered that several fuel assemblies contain secondary neutron sources that were not identified and analyzed as approved for storage in the DCSS. Criticality analyses are performed using the approved methodology by the DCSS vendor that account for the presence of these source pins, and it is determined that their presence is bounded by current criticality analyses. The 10 CFR 72.48 analysis of this change would show that it may be implemented without prior NRC review and approval, since no methodology changes

resulted, subcriticality of the DCSS with the new design was maintained, and no Technical Specification change was required.

Examples of Changes That Would Require Prior NRC Review and Approval

- A site-specific licensee assumes ownership of a DCSS and seeks to establish an in-house capability for performing criticality safety analyses for SNF to be stored in its DCSS. The performance of the criticality calculations using a different computational platform than that used by the DCSS vendor but using the same computational methodology results in a gain in margin (i.e., the results are not "essentially the same") for the limiting criticality case (e.g., calculated K_{eff} is reduced). This would be a non-conservative change, or a departure from a method that would require prior NRC review and approval under the requirements of 10 CFR 72.48.
- A DCSS vendor wishes to add a more reactive fuel type to the SNF to be stored, and compensate for the increased reactivity by increasing the minimum B^{10} content in the borated neutron absorbers. The increased B^{10} content results in a material configuration change that impacts the models used in the criticality analyses; the model change results in a non-conservative change in calculated K_{eff}. This would be a departure from a method that would require prior NRC review and approval under the requirements of 10 CFR 72.48.
- A DCSS vendor wishes to add a more reactive fuel type to the SNF to be stored, and compensate for the increased reactivity by removing the conservatism used in the assumptions for the model by accounting for fuel assembly hardware (grid spacers, end fittings, etc.) that was not accounted for in the approved and certified analysis. Although the criticality safety analyses reflect no change in K_{eff} for the new fuel type, K_{eff} for other fuel types would decrease, thus increasing the margin in a non-conservative direction; this change in assumption would constitute a departure from a method that would require prior NRC review and approval under the requirements of 10 CFR 72.48.

Fuel Specifications used in Criticality Analyses – Comparison with TS and Development of Guidance for STS Application

Based on Review of FuelSolutions W21 and W74, TN-32 and TN-68, HI-STORM 100 MPC-24 and MPC-68, and NAC-UMS; Documents reviewed were SARS (where available), TS from CoCs, SERs from NRC:

Current Technical Specifications Fuel Specifications:

1. Maximum Weight per Assembly (all)
2. Heat Load Limit per Assembly (all)
3. Cladding Material/Condition (all)
4. Initial Enrichment (all)
5. Burnup (FuelSolutions, TN-68, NAC-UMS, HI-STORM 100 as part of cooling table for Zr Clad, specific limit for HI-STORM 100 SS Clad)
6. Cooling Time
   a. Cobalt Content (FuelSolutions only)
   b. Storage Cask Dose Rate (FuelSolutions only)
   c. Canister Heat Load (kW/Canister, and kW/inch-Canister) (FuelSolutions only)
   d. Post-irradiation cooling time (HI-STORM 100, NAC-UMS; TN-68 and FuelSolutions has burnup vs. enrichment cooling table)
7. Fuel Assembly Type
   a. Max. Uranium Loading (all)
   b. Linear Uranium Loading (FuelSolutions only)
   c. Number of Fuel Rods (all)
   d. Minimum Clad Thickness (FuelSolutions, NAC-UMS; Not for TN-68; HI-STORM 100 has clad OD and clad ID specs)
   e. Minimum Pellet O.D. (Not for TN-68; HI-STORM 100, NAC-UMS has max pellet diameter)
   f. Rod Pitch (Max for TN-68, HI-STORM 100, NAC-UMS)
   g. Minimum Bottom Tie Plate Height (FuelSolutions only)
   h. Maximum Active Fuel Length (FuelSolutions, HI-STORM 100, NAC-UMS only)
   i. Maximum Individual Pin Enrichment (FuelSolutions, HI-STORM 100 MPC-68)
   j. Minimum Rod OD (TN-68, HI-STORM 100, NAC-UMS only)
   k. Minimum Enrichment (NAC-UMS only)
8. Preferential fuel loading (HI-STORM 100, NAC-UMS only)
9. Fuel Assembly Width (HI-STORM 100, NAC-UMS only)
10. Total Fuel Assembly Length (HI-STORM 100, NAC-UMS only)
11. Guide Tube Number and Thickness (HI-STORM 100 MPC-24 only)
12. Minimum guide tube thickness (NAC-MPC only)
13. Number of Water Rods, Water Rod Thickness (HI-STORM 100 MPC-68 only)
14. Channel Thickness (HI-STORM 100 MPC-68 only)

15. Storage of thimble plugs and burnable poison inserts (NAC-UMS)
16. Specific restrictions on stainless steel channels, unenriched fuel assembly storage in canisters (NAC-UMS only)

Technical Specifications Design Features for Criticality Control:

1. Flux Trap Size (HI-STORM 100 MPC-24 only)
2. Fuel Cell Pitch (HI-STORM 100 MPC-68, FuelSolutions)
3. ^{10}B loading in the Boral neutron absorbers (HI-STORM 100, FuelSolutions, NAC-UMS)
4. Minimum distance from base of canister to fuel region (NAC-UMS only)*not considered in NAC criticality analysis

Review of Cask Vendor Criticality Analyses

Key Criticality Analyses Features

1. Case studies to establish most reactive fuel type and average enrichment – Canister (W74) or Assembly Class (W21, TN, HI-STORM, NAC-UMS) – considers bounding values, as appropriate, from sensitivity studies for the following:
   a. clad material
   b. initial enrichment
   c. pellet stack UO_2 density
   d. number of fuel rods, including TN-68, HI-STORM 100 MPC-68) number of partial length rods
   e. clad O.D.
   f. clad thickness
   g. clad inner diameter
   h. pellet diameter
   i. fuel rod pitch
   j. active fuel length
   k. number of water holes
   l. number of non-corner water holes (W74 partial)
   m. number of inert rods (W74)
   n. bottom nozzle/tie plate height
   o. instrument tube and guide tube specifications (TN-32, FuelSolutions, HI-STORM 100)
   p. Fuel channels – presence and thickness/limiting thickness (HI-STORM 100 MPC-68, TN-68, NAC-UMS)
   q. Maximum loading in MTU (NAC-UMS)
2. Varying design conditions involving loading, closure, on-site transfer, and dry storage.
3. Treatment of special assembly classes (MOX, damaged in damaged fuel cans, partial, fuel debris). (FuelSolutions, HI-STORM 100 MPC-68F)

4. Analysis of mixed loadings of design basis fuel types and special assembly classes.
5. Full loading and partial loading analyses.
6. Demonstration that average enrichment is more conservative than multiple pin enrichments (BWRs) (FuelSolutions and TN-68). Also treatment of variable axial enrichment and partial length fuel rods (TN-68 only).
7. No credit for pellet dishing, fuel burnup, or fuel-related burnable neutron absorbers. (pellet dishing not mentioned for TN-32)
8. Conservatisms on borated neutron absorber plate material characteristics.
9. Shifting of fuel assembly positions radially to maximize system reactivity. (FuelSolutions, TN-32 and TN-68, NAC-UMS)
10. Loading of a single or multiple fuel assemblies with higher than design basis enrichment (FuelSolutions W21, TN-32 and TN-68)
11. Postulated reduction of pin pitch due to fuel grid crushing in a tipover accident. (TN-32 and TN-68)

Key Criticality Model Features

1. Normal (complete flooding with water [including pellet/clad gap] at a density providing optimum moderation, worst case [bounding] asymmetric assembly placement within basket guide tubes, application of worst case [bounding] material and fabrication tolerances) and Postulated Accident Conditions (normal plus bounding permanent deformation of guide tubes from hypothetical cask drop accident, axial detachment of guide tubes from basket structure, removal of transportation cask neutron shield assembly).
2. Sensitivity to basket design features (e.g., spacer plate axial spacing, spacer plate thickness, spacer plate hole location tolerances).
3. Worst case (bounding) configuration for canister mode (transfer, storage, transportation).
4. Bounding canister design for different basket configurations.
5. Bounding array – single package model vs. multiple package array.
6. Worst case multiple package array for maximum acceptable enrichment and design basis k_{eff} value.
7. Determination of most reactive configuration of fuel material in damaged fuel can and effect on loaded canister criticality.
8. Optimum, bounding array for most reactive partial fuel assembly. (FuelSolutions W74)
9. Bounding condition for criticality – normal vs. hypothetical accident
10. Use of an accepted code package (e.g., MCNP 4a, SCALE 4.3 CSAS, CASMO-3).
11. List of assumptions for development of analytical models (derived from FuelSolutions W21 SAR, HI-STORM 100, NAC-UMS lists for example).
12. Detailed description of analytical models (e.g., axial and radial models for normal and hypothetical accident conditions).

13. Use of reflected planes axially to prohibit neutron leakage. (FuelSolutions, TN-68, NAC-UMS)
14. Maximized conditions for fuel exposure in axial plane (W74 only).
15. Optimum pure water moderator density. (TN-32 in clad/pellet gap).
16. Optimum water density for borated (2300 ppm) water (TN-32 only).
17. Borated water draindown (TN-32 only).
18. Bounding fuel rod enrichment pattern.
19. Limiting USL value over range of parameter values, based on consideration of:
    a. assembly pin pitch
    b. enrichment
    c. water-to-fuel volume ratio
    d. H-to-^{235}U ratio
    e. B-10 concentration in separator plates (TN 32 and 68 only)
    f. Percentage of fissile material that is Pu (as opposed to U) (FuelSolutions only, for MOX fuel storage)
20. Accounting for differences between model and design (TN-32 and 68).
21. Determination of most reactive lattice (TN-68)
22. Preferential flooding/draining of fuel assemblies (FuelSolutions W74, TN-68)
23. Neutron source location, covergence (FuelSolutions, TN-68)
24. Water density sensitivity analyses – fuel array/canister array (FuelSolutions, NAC-UMS)

5.1.2 CASK Loading, Unloading, and Preparation Program

A program shall be established to implement the SAR requirements for loading fuel and components into a CASK, unloading fuel and components from a CASK, and preparing a CASK for storage. The requirements of the program for loading and preparing a CASK shall be met prior to declaring a CASK in storage.

The program shall address the following requirements as a minimum:

a. Dissolved boron concentration in the CASK cavity, if applicable, including requirements for independent measurements;

b. Qualification of fuel assemblies for loading (e.g., cladding material/condition, burnup, cooling time) and verification of correct loading;

c. Removable contamination acceptance criteria on components to be transferred to storage;

d. Canister vacuum acceptance criteria and vacuum drying time limits;

e. Helium backfill acceptance criteria;

f. Leak rate testing acceptance criteria;

g. Cask dose rate measurements and acceptance criteria;

h. Cask unloading acceptance criteria;

i. Time limit for transfer to a storage cask from a transfer cask, as applicable.

5.1.3 On-Site Transportation Program

A program shall be established to implement the SAR requirements for on-site transportation of the CASK. The requirements of the program shall be met prior to declaring a CASK in storage.

The program shall address the following requirements as a minimum:

a. Time limit for transfer to a storage cask from a transfer cask, as applicable;

b. Removable contamination acceptance criteria on components to be transferred to storage;

c. Limitations on fuel supply for surrounding vehicles;

d. Limitations on CASK temperatures;

e. Limitations on transportation roadway drop-offs or maximum lifting heights;

f. ISFSI pad requirements to interface with CASK design requirements;

g. CASK spacing requirements for ISFSI pad storage;

h. Radiation dose survey requirements.

5.1.4 CASK Storage Integrity Program

A program shall be established to implement the SAR requirements for periodic monitoring of CASK storage integrity.

The program shall address the following requirements as a minimum:

a. Time limitations for vents blocked loss-of-cooling;

b. Surveillance requirements for air inlet and outlet openings;

c. Surveillance requirements for CASK temperature monitoring;

d. [Surveillance requirements for CASK seal monitoring].