

**Issue 1:** Level of Detail in FSAR

**Summary of Issue**

SFST staff personnel are requiring a significant increase in the level of detail in FSAR Chapter 5 (Shielding Evaluation) and Chapter 10 (Radiation Protection) from applicants as amendment requests for storage system designs are reviewed. In many cases, vendors are being requested to add detail to the FSAR for CoC amendments that was not required for the original certification and is, in many cases, beyond what is reasonable or necessary for the shielding reviewer to make a safety judgment. In addition, calculation packages and input/output files are routinely requested by the reviewers. Submittal of calculation packages is not required by the regulations and input/output files for computer models are not appropriate information to be included in the FSAR. These staff requests seem to be excessive and not consistent with the concept of an audit review of the FSAR information.

**Proposed Resolution**

FSAR Chapter 5 should include the information used to determine a representative sample dose rate for the transfer cask and the storage cask or horizontal module containing design basis fuel. If different cask models provide significantly different shielding (e.g., different transfer casks), the expected difference in dose rates should be discussed and a separate analysis performed as necessary. Input parameters for the shielding analysis should be nominal values that are realistically conservative, but not necessarily absolutely bounding for every possible combination of allowable contents and component model. Specifically, we recommend that SAR Chapter 5 contain the following:

- Figures and a summary description of the component design features important for shielding, including nominal dimensions.
- A description of the storage system shielding model(s) including the material of each cask component or subcomponent used in the shielding analysis. A detailed elemental breakdown of standard materials (e.g., steel, concrete, and lead) should not be required. Elemental descriptions of non-standard materials, such as polymer-based neutron shields should be provided.
- A description of the physical model of the fuel assemblies and other allowed contents, such as non-fuel hardware.
- A description of the source term for each burnup/cooling time case presented in the FSAR, how it was developed, and how it represents a reasonably conservative and representative sample case for the allowed contents.

- A description of the analytical method used, including the computer codes, editions, and versions.
- A list or table containing key inputs, significant assumptions, and sources of conservatism.
- Results of the shielding analysis for the transfer cask and the storage cask or module, including dose versus distance results for a single cask and a representative ISFSI array.

FSAR Chapter 10 should include a summary discussion of storage system design features and operating recommendations that are directed toward minimizing dose during loading and unloading operations consistent with good ALARA practices. This should include a clear distinction between mandatory and optional temporary shielding measures. In addition, a single estimate of occupational exposure for loading the system should be provided, using the dose rates in the shielding analysis. A sample of the level of detail that should be used in a Chapter 10 table of estimated dose for loading operations is provided below for a typical canister-based storage system. The estimated occupational doses in the FSAR should be determined based on the vendor's best judgment using the standard procedures in the FSAR and reasonable estimates for stay times, personnel locations in the vicinity of the source, and crew size. The information may be varied as appropriate for bare fuel casks or unique operations for a particular dry storage system. The mathematical estimation of doses using the details of operation and number of personnel required should be retained in a supporting calculation package, available for inspection by the NRC.

ACTIVITY	TOTAL ESTIMATED DOSE (person-mrem)
Load fuel into cask and install lid	
Remove loaded cask from pool and move to preparation area	
Weld canister lid and perform inspections	
Canister pressure testing	
Drain, dry, and helium backfill canister	
Weld inner port covers	
Helium leak testing	
Final welding	
Install transfer cask lid	
Move transfer cask to truck bay	
Transfer canister to vertical storage cask	
Move vertical storage cask or transfer cask to ISFSI	
Transfer canister to horizontal storage module	

## Basis

Cask users (mostly Part 50 licensees) maintain extensive radiation protection programs designed to address handling and processing radioactive materials, including very high dose rate items such as fuel assemblies and primary coolant filters, with the overarching goal of keeping personnel doses ALARA. Regulatory Guide 8.8 guides ALARA implementation at the plants. Plant radiation protection programs and the radiation work permits (RWPs) controlling the field work are designed to deal with radioactive materials *as they are* in the plant, not as they are reported in calculations or licensing basis documents that, necessarily, over-estimate the dose rates by maximizing the potential source term. Thus, temporary shielding and other personnel protective measures not mandated by the storage system vendor are determined by the plant staff in the development of the RWPs based on actual conditions in the plant, not the information in the storage system FSAR or supporting calculations.

Using actual dose rates rather than overly conservative estimates in developing the RWPs is an important ALARA concept because licensees must balance the estimated dose for installing temporary shielding against the estimated dose for performing the activity without the temporary shielding to determine the appropriate, ALARA-based plan of action. At most, the information in the storage system FSAR is used by plant radiation protection personnel as a starting point that provides a conservatively high value for dose rates, not the expected value. They may also use the FSAR dose rates versus distance as a conservative way to estimate offsite exposures for documentation in the 72.212 Report.

The level of precision required of the shielding analysis and associated FSAR detail should reflect the use of the information by plant personnel as a starting point in their development of RWPs and establishing temporary shielding needs. The dose rates reported in the FSAR, as recommended herein, are sufficient for the NRC staff to make a judgment on the cask's ability to be deployed and meet site-specific regulatory requirements for off-site dose (10 CFR 72.104) and occupational exposure (10 CFR 20) as demonstrated by the user. It is not consequential to the user if the cask/module surface dose rate reported in the FSAR for design basis fuel is estimated to be 300 mrem/hr or 350 mrem/hr at a specific location. But it is helpful to know whether the surface dose rate with design basis fuel is estimated to be in the range of 300 mrem/hr or 3,000 mrem/hr (i.e., the order of magnitude). The level of detail in the FSAR and associated staff review should reflect the manner in which the information contained in FSAR Chapter 5 is used at the plants.

With regard to FSAR Chapter 10, the storage system vendors have no control over the exact details of the site-specific implementation procedures, which reflect real plant needs, equipment and capabilities. Personnel stay times, locations, and crew sizes are also determined plant-specifically and change as plants gain experience working with the storage system. Thus, having a great amount of detail or precision in Chapter 10 dose estimates yields very little insight into the doses that will be experienced in the field or

the capability to reduce doses in a given plant. As discussed in the November meeting, licensees monitor accumulated dose closely and are constantly looking to reduce overall dose for cask loading operations as part of their ALARA programs, irrespective of the information in the storage system FSAR.

**Issue 2:** Bounding Calculations and Results

**Summary of Issue**

Shielding calculations are performed with numerous inputs reflecting the storage system physical design, fuel and non-fuel hardware physical design, and the source term of the contents. A single certified storage system can be comprised of a number of different storage cask, canister, and/or transfer cask models (some site-specific) that may or may not produce significant differences in the shielding analysis results. As described in Issue 1, only a calculated dose rate from a reasonable conservative, representative model is required for the NRC staff to conclude that reasonable assurance has been established that the storage system will allow the licensee user to meet the dose limits in 10 CFR 72.104 and 10 CFR 20.

Proposed amendments to CoCs that introduce new allowable contents and/or storage system components may be similar to or bounded by the previously approved analysis. As such, new analyses may not be required for each amendment request that adds contents to the storage system or changes the shielding design.

**Proposed Resolution**

NRC staff should recognize their previous approvals of shielding work in their review of amendment requests if the storage system designer has shown the previous work to provide reasonable assurance that the regulations can be met with the proposed changes. For example, if an amendment request adds fuel that is of a lower or similar source term than previously reviewed and approved, the review should simply be an audit of the source term and shielding model, and conclude that it is acceptable by comparison to that previously approved. If the storage system designer chooses to perform a full shielding analysis and the resulting dose rates are lower than previously approved, the analysis should be reviewed for reasonableness and accepted based on the previous approval of the higher dose rate results.

**Basis**

There are no regulatory limits on the contact dose rate from a transfer cask or storage cask/module. If the NRC staff has previously accepted a higher calculated dose rate, then a lower calculated dose rate for the same component should be acceptable. Questions from the staff on the shielding evaluation should be limited to new information and only that information needed to confirm that the previous analysis is representative for the change requested. The level of detail of the NRC review should vary based on what has been previously approved and what the changes are.

NRC staff reviews are audit reviews of the material in the storage system FSAR, not quality assurance-level verifications of the supporting calculations. The storage system vendors are responsible for preparing and verifying the accuracy of the supporting

calculations under their NRC-approved QA programs. The NRC staff is responsible for making a safety judgment based on reasonable assurance given the information presented in the FSAR. This distinction is important for an effective, efficient regulatory process.

**Issue 3:** Definition of a Small Change versus a Small Effect

**Summary of Issue**

Shielding calculations are performed with numerous inputs reflecting the physical design of the storage system and allowed contents and the source term of the contents. NRC shielding reviewers are currently requesting a level of precision in the shielding analysis input values that is inappropriately high and not commensurate with the safety significance of the results or regulatory requirements. In particular, the following inputs to the shielding analyses are receiving an inordinate amount of scrutiny by the NRC staff given the inconsequential effects of small changes to these inputs:

- Fuel axial burnup distribution
- Fuel axial enrichment distribution
- Damaged/failed fuel
- Isotopic composition of the fuel and fuel assembly hardware in the shielding model
- Axial uranium blankets

**Proposed Resolution**

The shielding review conducted by the NRC staff on these inputs should be a “reasonableness” review. The following approach for each of the input items listed above should be allowed for storage system shielding analyses:

**Axial Burnup Distribution**

A representative axial burnup distribution for PWR and BWR fuel should be used based on the technical literature.

**Axial Enrichment Distribution**

A uniform assembly average enrichment distribution should be assumed over the active fuel length of the assembly.

**Damaged/Failed Fuel**

Damaged/failed fuel does not need to be uniquely modeled or otherwise considered differently than undamaged or intact fuel.

Isotopic Composition of the Fuel and Fuel Assembly Hardware in the Shielding Model

The fuel assembly in the shielding model should be represented as a fresh assembly. (The source terms are calculated considering burned assemblies.) Standard material compositions may be assumed for other fuel hardware, such as the fuel cladding and fuel skeleton.

Axial Uranium Blankets

The natural uranium blanket at the top and bottom of the fuel assemblies may be modeled as enriched zones.

Basis

It is commonly understood by storage system shielding analysts that variations in the above items in the shielding model have no consequential effect on the calculated dose rates. Shielding analysis inputs and assumptions pertaining to each of the above items have been reviewed and approved numerous times over the years with many RAIs issued and answered to reinforce this fact. Consistently it has been shown by the storage system vendors that variations in these items have virtually no effect on dose rates at the ISFSI controlled area boundary, where the regulatory limits in 10 CFR72 apply, and only modest effects on dose rates on or near the transfer cask, storage cask or horizontal storage module.

Therefore, refinements in the above mentioned parameters will not enhance the ability of the vendorsto demonstrate in the FSAR that the storage system is capable of meeting the regulatory dose limits at the controlled area boundary. In addition, refinements in these parameters will not increase the usefulness of the calculated dose rates when, as an example, it is not consequential to the user if the cask surface dose rate reported in the FSAR for design basis fuel is estimated to be 300 mrem/hr or 350 mrem/hr at a specific location. Therefore, these inputs should be nominal, reasonable and based on sound science and applicable codes and standards or technical literature and the NRC should review them in that light.



**Issue 4:** Burnup and Cooling Time versus Heat Load Limits in the CoC

**Summary of Issue**

The NRC staff requires storage system designers to include fuel and non-fuel hardware burnup versus cooling time limits in addition to decay heat limits for each fuel storage location in the CoC technical specifications. This is redundant and unnecessary.

**Proposed Resolution**

Permit vendors to specify either a decay heat limit or a burnup versus cooling time limit for the contents of each fuel storage location, but not necessarily both. The limits should be based on the storage system thermal analysis, not the shielding analysis.

**Basis**

The storage system designs are driven primarily by maximum heat load. Thus, the limits on contents are based on the governing thermal analysis. CoCs are currently required to include decay heat limits and burnup/cooling time limits for each storage location (i.e., fuel cell). Decay heat limits are chosen to ensure the total heat load limit for the cask is not exceeded and, in some cases, to accommodate zoned fuel loading. The fuel cladding is protected against gross rupture by the users' compliance with the CoC limits on decay heat load in each fuel storage location. These limits are based on the supporting thermal analysis, which ensures the fuel cladding temperature in all locations will remain less than the applicable limit.

Vendors recognize that variations in fuel configuration as well as initial enrichment and burnup history affect source terms and resulting dose rates to a minor extent. However, because there are no regulatory limits for dose rates on contact with the spent fuel storage system components, there is no basis for choosing burnup and cooling time limits based on the shielding analysis. Therefore, the vendors currently determine the allowable burnup and cooling time combinations for the CoC based on the allowable heat load. This results in unnecessary and redundant requirements in the CoC.

**Issue 5:** Technical Specification Dose Rates

**Summary of Issue**

The NRC staff requires storage system designers to include in the CoC technical specifications maximum cask/module dose rates, locations for measuring those dose rates, and, in some cases, specific times during cask loading operations for measuring the dose rates. These Technical Specifications are unnecessary to assure safety.

**Proposed Resolution**

Do not require transfer cask, storage cask, or horizontal storage module dose rate limits to be included in the CoC or technical specifications. Estimated, representative dose rates from these components should be included in the storage system FSAR.

**Basis**

Dose rates will be reported in the storage system FSAR and will be significantly above those actually measured in the field. Licensees measure dose rates on the storage system components at appropriate times in the loading process, and have an expectation on the order of magnitude of the dose rates based on the FSAR information. They will take suitable corrective actions if unexpectedly high dose rates are measured, simply because the site radiation protection programs demand such prudence.

Historically, NRC has believed that having a dose rate limit in the CoC may detect a cask misloading. It would be extremely unlikely for a dose rate measurement to detect a misloading event. Exceeding a CoC dose rate limit would only reveal the most serious of misloading events (i.e., multiple over-burned assemblies in the peripheral fuel cell locations). Most misloading events, because they involve only a few fuel assemblies either cooled slightly too little or burned slightly too long would never be discovered by a dose rate reading exceeding a CoC limit. This is because the CoC dose rate limits (which are based on the FSAR shielding analysis) are conservatively high and represent the full spectrum of allowed contents for the storage system. A misloading event of any severity occurring anywhere other than in the peripheral fuel cells would not likely be detected by elevated dose rates due to the significant self-shielding provided by the outer fuel assemblies. No actual misloading events that have occurred, to our knowledge, have been discovered via dose rate measurement.

As discussed at the November meeting, plants track the contents loaded into fuel casks very closely, including independent verification and, in many cases, video records. This attention to detail in cask loading is driven by the requirements to control special nuclear material in the Part 50, Part 72, and Part 74 regulations. Misloadings are taken very seriously and are addressed in the licensee's corrective action program to determine the cause and prevent recurrence. The cause and corrective action information is shared

either formally (e.g., via event notification) or informally (e.g., via storage system users groups) to help reduce the likelihood of a similar event elsewhere.

**Issue 6:** Computer Codes

**Summary of Issue**

Storage system designers use a number of computer codes to develop the source terms and perform the shielding analyses that calculate the dose rates for the storage system. System analysts generally use the versions of computer codes available at the time of original certification and validated for use in their quality assurance program. As time goes by, the previously used codes may be revised with updated information, such as new features, or may no longer be supported by the code developers. NRC staff reviewers often expect vendors to adopt later computer code versions without demonstrating that there is a significant safety benefit to be gained.

**Proposed Resolution**

Storage system owners may use any computer code and version determined to be suited for the problem being analyzed provided any safety-significant issues with the code have been identified and addressed appropriately. The same codes and versions approved for use in the original CoC may be used, at the vendor's discretion, to support subsequent amendment requests. Updates to later version or new codes should be at the vendor's discretion.

10 CFR 21 should be used as the vehicle to drive mandatory use of new or updated computer codes.

**Basis**

Updating to a different or newer computer code or version is a burdensome and costly undertaking for the vendors and should only be required if there is a commensurate safety improvement. Unless a "safety-significant" issue has been identified with the computer code version used in the analysis described in the FSAR, the dose rates calculated using this version are still accurate, even if a newer version of the code produces slightly different results. The issues found and changes made to the computer programs used for shielding are typically not safety-significant. That is, they improve the precision of the results but do not cause a significant enough increase in the dose rates reported in the FSAR to be of concern to storage system shielding analysts, the NRC, or licensee users.

The NRC should not be requesting vendors to update computer programs on an ad-hoc basis solely when a CoC holder submits an amendment application. The NRC's regulations at 10 CFR 21 are in place to identify defects in basic components that create a substantial safety hazard. Defects, in this context, include errors in computer codes used for important-to-safety work. If the users are NRC licensees or certificate holders, they are obligated to review the error for impact on their work and report it to the NRC if a substantial safety hazard is created.

As a parallel example, nuclear power plant Emergency Core Cooling Systems (ECCS) are designed and validated using thermal-hydraulic computer codes (e.g., RELAP) that are subsequently revised on a periodic basis by the code developers. The plant owners are under no obligation to update their ECCS analyses using the revised code unless a safety-significant issue is found in the code version used, in which case the NRC would use the generic issue and backfit processes to impose the new requirements. 10 CFR 21 would likely be the vehicle used to identify such issues with the computer code.