HLWYM HEmails

From: Sheena Whaley
Sent: Thursday, September 13, 2007 1:12 PM
To: Christopher Ryder; Robert Johnson (NMSS)
Cc: Oleg Povetko; Banad Jagannath; David Dancer; Rosemary Reeves; Tae Ahn
Subject: input
Attachments: Phase II B Sheena comments 9_13.wpd

Chris,

Here is my revised input. I didn't get to put in the info from the runs I did but if you have time, maybe you can graph the results (see below) and stick it in the section on neutron poisons. Otherwise, just take out the note that I had left myself to put that in there. I really should re-run these, but I think Oleg checked my inputs and outputs so hopefully they looked ok.

Note that I would also want the text below included to explain that this was for a very specific case, etc especially since calcs done by Oleg for a different cask configuration indicate very different results.

Oleg, if you want an electronic version of any of the references I listed, let me know and I'll put them on the share drive once I get back. I'll be in Oak Ridge next week.

keff with Different Boral Plates Missing
Basis: a PWR cask with 24 assemblies, boral neutron poison and flux traps; a TAD does not have flux traps, uses borated stainless steel and only holds 21 PWR assemblies.
Fuel: fresh (4.2 wt percent), 15 × 15 Westinghouse assembly

keff with all boral plates present	0.9441 + 0.0016
keff with two boral plates missing	0.9439 + 0.0018
keff with four boral plates missing	0.9458 + 0.0015

Preliminary calculations performed by the NRC staff show the change in keff with up to four neutron absorber plates (boral plates) missing. This true ONLY for a PWR canister with no flux traps. These canisters may be unloaded at the GROA. However, DOE will be loading into a TAD canister with no flux traps and neutronically it's quite a change. Also the effects for a BWR canister should be evaluated.

Sheena

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Subject: input
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From: Sheena Whaley

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Files	**Size**	**Date & Time**
MESSAGE	1643	9/13/2007 1:12:00 PM
Phase II B Sheena comments 9_13.wpd	1702825	

Options
Priority: Standard
Return Notification: No
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8 Criticality Insights for handling Spent Nuclear Fuel

8.1 Background

Part 63 requires DOE to prevent and control a criticality. DOE has indicated that it will demonstrate that a criticality is not credible (beyond Category 2) for preclosure operations. To do so, DOE will analyze both normal operations, as well as scenarios for category 1 and category 2 event sequences.

At the GROA, DOE will handle a wide variety of spent nuclear fuel assemblies (SNF) and canisters. The GROA is expected to handle approximately 122 different types of commercial spent nuclear fuel (CSNF) assemblies, several types of DOE spent fuel assemblies, vitrified high level waste, and Naval fuel, described in Section 2. In contrast, at nuclear power plants the assemblies are usually the same type and the canister types are limited. However, the SNF handling operations at the GROA are expected to be very similar to operations performed under Parts 50, 71, and 72.

DOE will open and repackage CSNF not shipped in TAD canisters in the wet handling facility (WHF). The WHF will therefore process many different cask and canister designs that are constructed of a variety of materials and basket configurations. Other DOE SNF and HLW will be handled in canisters only. These will be shipped in canisters that are approved for disposal, that only require removal from the transportation overpack and placement into an aging or disposal overpack. The GROA facilities other than the WHF will handle a limited number of canister and cask designs. For all of these operations, since casks and canisters are typically designed to store more than one type and configuration of CSNF, and many cask and canister types will be handled at the GROA, the applicant should demonstrate that criticality requirements are satisfied for the bounding case.

In general, no one type of CSNF assembly or cask/canister will reasonably bound all criticality scenarios. The applicant should justify which assemblies reasonably bound each system (canister and fuel assembly), especially if the system's calculated keff is near the upper subcritical limit, or a small change to a parameter can result in a large change in keff. This also applies to burnup credit; the most reactive unirradiated assembly is not necessarily the most reactive after irradiation.

A determination of which fuel is bounding in a criticality analysis depends on many factors and usually requires examination of several types of fuel assemblies and compositions. The design-basis fuel has often been the Westinghouse 17x17 optimized fuel assembly (OFA); however, this will not be the case for all event sequences effects on reactivity due to different materials, etc. Also, the most bounding fresh CSNF assembly is not necessarily the most reactive after irradiation. Therefore, the reviewer should verify that the fuel assembly used as the design basis is the most reactive for the specific scenario.

Some of the more important parameters to consider are:

- Type of assembly to include vendor, lattice size, and enrichment (e.g., Westinghouse PWR 15x15, 5 wt% U-235)
- Burnup
- Cooling time
- Condition of fuel
- Non-fuel components
- Neutron poisons (soluble and fixed)

Moderation
Basket or SFP rack configuration and materials, and number of assemblies

Additionally, the reviewer needs to clearly understand the limitations of the codes and data used to determine criticality safety.

8.2 Parameters Important to Criticality Safety

8.2.1. Type of Assembly

The type of CSNF assembly and maximum fuel enrichment should be specified and used in the criticality calculations (reference 1). Justification of the bounding assembly type is necessary because sometimes the bounding assembly type for the normal or nominal scenarios is not the most reactive during accident scenarios, even for the same cask or canister type. However, if the all the assemblies are of a similar lattice size (e.g., 15x15) and enrichment, then evaluations for CSNF have shown that the change in reactivity is usually negligible. The criticality evaluation should be based on the highest enrichment (if fresh fuel is assumed) for a given assembly type, unless loading controls prevent the canister from being completely loaded with the maximum enrichment.

Some boiling water reactors (BWR) use assemblies with multiple fuel pin enrichments. For these, the criticality evaluation should use the maximum fuel pin enrichment present (reference 1). Depending upon the design, the use of assembly averaged, or lattice averaged enrichments may be used if it is demonstrated that this produces conservative calculational results. Because of the natural uranium blankets present in many BWR designs, use of an assembly-averaged enrichment is not normally considered appropriate or conservative for BWR fuel.

8.2.2. Burnup
DOE has indicated that it may rely on burnup credit (29 nuclides) to screen out a criticality accident, including burnup credit for BWR fuel. Current NRC guidance and licensing precedence provide for partial burnup credit for PWR fuel only due in part to the lack of relevant experimental data necessary to adequately validate calculated isotopic concentrations and cross sections (reference 2). One reason for limited burnup credit is due to the lack of radiochemical data for many of the nuclides, especially the fission products. Chemical assay data have historically focused on the major actinides within PWR spent fuel. DOE has proposed to obtain the technical data needed and is discussed in detail in the Technical Work Plan for Development of Technical Data Needed to Justify Full Burnup Credit in Criticality Safety Licensing Analyses Involving Commerical Spent Nuclear Fuel, TWP-EBS-MD-000019 Rev 01, February 2007.

There are also no benchmark critical experiments with CSNF in a cask-like environment, so DOE has proposed using commercial reactor critical experiments to perform validation. To do so, DOE will need to demonstrate that the experiments are representative of the cask system and that the bias and uncertainty can be determined. Reference 5 provides information on those nuclides, both actinides and fission products, that are important to the neutron absorption rate. has . Table x, taken from reference 5, gives the fractional contributions and rankings of actinides in the PWR fuel are given for decay times of **5** years.

The range of realistic reactor operating conditions also needs to be known for the depletion analysis. Operating parameters that need to be addressed are summarized in Table 4 of reference 3 and include fuel temperature, moderator temperature, moderator density, soluble

boron concentration, operating history, specific power, and use of burnable neutron absorbers. The cooling time is also important and is discussed below. Another important consideration is the assembly burnup profile. Reactor operations result in non-uniform axial burnup profiles. Typically, a realistic axial profile gives a higher cask keff for burnups greater than 20 GWD/MTU than does a uniform axial profile assumption. This is because of the difference between assembly averaged burnup and the burnup in the end region increases with assembly burnup. In a BWR, the burnup profile is further complicated because of (1) varying moderator density, (2) varying fuel enrichments, (3) axially varying poison rod enrichments, and (4) partial control rod insertion. Horizontal burnup profiles should also be considered but due tp fuel management practices, any horizontal burnup gradient in normal discharged fuel is minimized (reference 3).

Finally it is important to note that the most bounding fresh CSNF assembly is not necessarily the most reactive after irradiation.

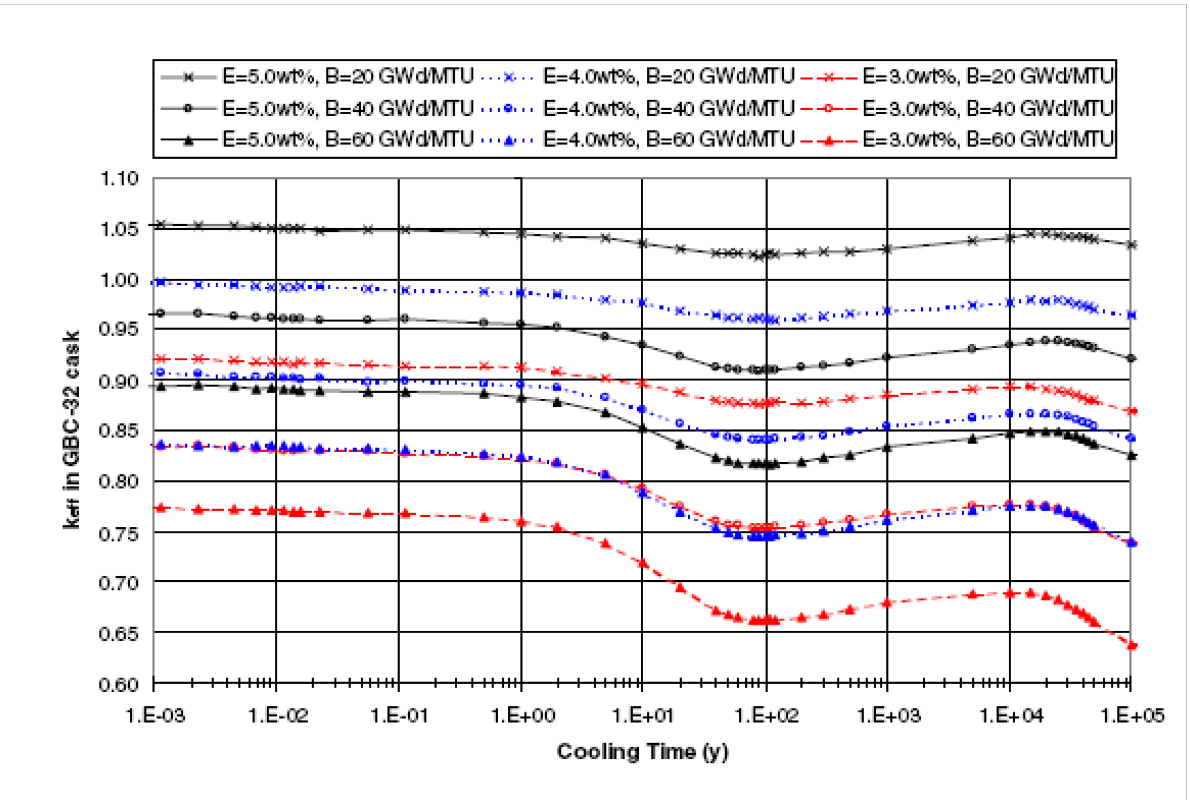
Table x Criticality-safety rankings for dominant actinides in PWR fuel (taken from reference 5)

	Burnup / enrichment											
	20 GWd/t		50 GWd/t		30 GWd/t		60 GWdIt		40 GWd/t		70 GWd/t	
	3wt%		3wt%		4wt%		4wt%		5wt%		5wt%	
Nuclide	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5 year cooling												
Am-241	0.63	7	1.36	6	0.82	7	1.4	6	0.95	7	1.43	6
Am-243	0.03	12	0.50	10	0.08	12	0.56	10	0.13	12	0.61	11
Np-237	0.26	8	0.73	9	0.42	8	0.91	9	0.59	8	1.08	9
Pu-238	0.05	11	0.41	11	0.10	11	0.52	11	0.17	10	0.63	10
Pu-239	25.23	2	27.66	2	24.71	2	26.66	2	24.14	2	25.82	2
Pu-240	6.48	4	9.17	3	6.87	4	8.97	3	7.04	4	8.80	3
Pu-241	3.06	5	6.51	4	3.70	5	6.39	4	4.08	5	6.30	5
Pu-242	0.18	9	-1.05	7	0.32	9	1.09	8	0.44	9	1.13	8
U-234	0.14	10	0.08	12	0.15	10	0.10	12	0.16	11	0.12	12
U-235	22.88	3	5.22	5	21.69	3	6.31	5	20.97	3	7.15	4
U-236	0.75	6	0.94	8	1.01	6	1.17	7	1.23	6	1.37	7
U-238	30.88	1	30.35	1	29.27	1	29.09	1	28.00	1	28.00	1
Total	90.59		84.12		89.17		83.34		87.94		82.62	

8.2.3. Cooling Time

When SNF is assumed to be unirradiated, then cooling time (time after discharge from the

reactor) is only important for decay heat and radiation source terms. When burnup credit is taken, cooling time is an important consideration for criticality safety. After SNF is discharged from a reactor, the reactivity continues to vary as a function of time due to the decay of unstable isotopes. Fuel discharged from a reactor increases in reactivity for several days due to the decay of short-lived poisons. After this initial increase, reactivity decreases with time until about 100 years (reference 6). The reactivity then increases, peaking a second time at around 30,000 years. The reactivity of this second peak is always less than that occurring at 5 years. As burnup increases, the effect of cooling time is more pronounced due to the increased quantity of 241 Pu and fission products relative to the remaining inventory. Reference 6 studied the effect of cooling time on reactivity for various initial enrichments, burnups, and selected nuclide sets. Figure x, taken from reference 6 shows the effect of cooling time on *keff* values for various burnup and initial enrichment combinations for actinide + fission products, which represents what DOE is currently proposing.The *keff* values vary as a function of cooling time due to the decay of unstable isotopes andsubsequent buildup of others. The concentration of unstable isotopes is dependent upon the dischargeburnup. The effect of cooling time is shown to increase with burnup.



Reactivity behavior in the GBC-32 cask for actinide + fission product burnup credit as a function of cooling time for various initial enrichment and burnup combinations. The calculations include an axial burnup distribution (from reference 6).

8.2.4. Condition of Fuel

DOE has provided only limited information on how it will model fuel in damaged fuel containers, which is not hypothetical and must be addressed. To date, casks have not been analyzed for fuel that is significantly damaged or has a gross cladding defect. Several transportation SARs

reviewed by the NRC show that damaged fuel has the potential to be more reactive than intact fuel. Consequently, criticality analyses have generally specified that any damaged fuel rods should be replaced with dummy rods that can displace an equal amount of water as the original rods, or damaged fuel is placed into cans before loading into a canister. During transportation, fuel that will be repackaged at the GROA may become damaged.

8.2.5. Non-fuel Components

The handling/aging/disposal of non-fuel core components in a cask, should be considered as these can affect the reactivity of the system. These displace water, and where borated water is used, they also displace the neutron absorber.

8.2.6. Neutron Poisons

The criticality design often relies on neutron poisons. These may be in the form of fixed poisons in the basket structure and/or soluble poisons in the water of the spent fuel pool. If a minimum boron content for fixed poisons is specified, then the reviewer should verify the minimum required boron concentration. Reliance is usually placed on the qualification and acceptance testing, especially for new materials and manufacturers. ASTM C 1671 – 07, Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging, provides guidance on testing of certain neutron absorber materials.

The criticality analysis should not take credit for all of the boron because of the various material properties. Typically 75-90% credit is given, depending on the material. Due to neutron streaming in Boral, this material is typically only given 75% credit. Boral is used in some canisters that may be unloaded at the WHF. DOE has proposed to use borated stainless steel in the TAD canisters, and nickel-gadolinium in the DOE spent fuel canisters. Other neutron poisons such as hafnium rods may be used.

Also, when solid neutron-absorbing materials are used, a positive means to verify their continued efficacy should be provided. Continued efficacy can be demonstrated by showing that the neutron flux from the irradiated fuel results in a negligible depletion of poison material over the required performance period. If relied upon the structural integrity and potential for poison material degradation during storage should be evaluated. Staff performed some calculations using a generic canister where boral plates were removed to determine the reactivity. (FINISH)
(REFERENCE REPORT ON MISSING BORON)

If borated water is used for criticality control, administrative controls and/or design features should be implemented to ensure that accidental flooding with unborated water cannot occur, or the criticality evaluation should consider accidental flooding with unborated water.

8.2.7. Moderation

In addition to a fully flooded cask, configurations in which the cask is partially filled with water (borated, if applicable) and the remainder of the cask is filled with steam consisting of ordinary water at partial density should be considered, if applicable. These configurations are generally considered to be representative of loading and unloading operations in the spent fuel pool. The

possibility of preferential or uneven flooding within the cask, if such a scenario is credible for the given cask design (e.g., because of blockage in small flow or drain paths) should be considered.

For analyses of a cask model with separate regions of water and steam, the use of a multigroup cross-section set raises additional concerns. Differences of the flux spectra in the two regions must be addressed. If the results of these calculations indicate that keff is close to 0.95, additional independent calculations using a different code and/or cross-section library may be helpful.

8.2.8. Basket or SFP rack configuration and materials, and number of assemblies

Deviations from nominal design configurations should be evaluated. The evaluation of keff should not be limited to a model in which all of the fuel bundles are neatly centered in each basket compartment. For example, a cask with steel confinement and lead shielding may have a higher keff when the basket and fuel assemblies are positioned as close as possible to the lead (reference 1).

8.3. Computer Programs and Data

8.3.1. Computer Programs
Both Monte Carlo and deterministic computer codes may be used fo generally better suited to three-dimensional geometry the Monte Carlo codes are more widely used to evaluate SNF and canisters. Two frequently used Monte Carlo codes are SCALE/KENO and MCNP. KENO is a multigroup code that is part of the SCALE sequence, while MCNP permits the use of continuous cross-sections. If a multigroup treatment is used, ensure that the neutron spectrum of the system is considered. Also verify that the cross-section set is collapsed with an appropriate flux spectrum. Some cross-section sets include data for fissile and fertile nuclides (based on a potential scattering cross-section) that can be input by the user. If the applicant has used a stand-alone version of KENO, ensure that potential scattering has been properly considered.

8.3.2. Data
A reviewer needs to understand the limitations of the cross section library used. In general, all SCALE libraries perform acceptably for unmoderated fast systems, and hydrogen moderated thermal systems. SCALE libraries have generally shown a negative bias of 1-2% for low enriched, water moderated systems (reference 4).

For multigroup calculations, the neutron flux spectrum used to construct the group cross-sections should be similar to that of the system modeled. Factors that affect the performance are: adequacy of the data used to generate the cross-section libraries, number of energy groups, location of energy group boundaries, flux used, and methods used to self-shield resonance nuclides. For example, since SCALE 218 and 238 group libraries were generated with different flux functions (at least for up to ENDF/B-IV), they can calculate differently for different classes of systems, especially where nonresonance nuclides play a role in absorption or transport (Al, Si, S, Cl, K).

The application of cross sections outside the range of validation should be done with care due to the potential for unidentified deficiencies in both the data and the manner in which the cross sections are processed. Table 8.1 of reference 4 provides a summary of SCALE library performance for different systems (thermal, fast, etc.).

An example of some known problems that may be applicable to the preclosure area are:

Use of hafnium data and gadolinium data (possible control rod or neutron absorber material in some fuel) in the ENDF/B-IV libraries used with the SCALE 218- and 27- group libraries is not recommended because the hafnium data was not included and there were problems with the resonance data for gadolinium.

The ENDF/B-IV libraries are not recommended for Pu systems (will handle Pu and MOX fuel), especially in the intermediate spectrum due to poor performance.

The 218 ENDF/BIV library performs very different for fast vs. thermal systems with U-233, thus should be carefully evaluated over the appropriate energy range.

8.4 Post Closure Issues

DOE probability arguments depend to a significant degree on the criticality design criteria. As DOE has chosen some novel (untested) design criteria and analytic approaches that in part ignore existing guidance, the review of DOE's design may be complex and resource intensive. The effects of criticality events were evaluated in a sensitivity study using TPA 4.1j. However, limited technical basis was provided for the parameter ranges used in the sensitivity study.

Critical events produce several unique conditions (though not necessarily important) that need to be understood:

- changes to the inventory to the fuel,
- radiolysis, in particular neutron irradiation, because it may affect the bulk chemistry inside the waste package
- long-term elevated temperatures
- mechanical Disruption of the Fuel and/or the Waste Package because Energetic criticality events have the potential to deliver mechanical energy to the waste package and disrupt the fuel.

8.5 Review Insights

Section x (page 33). Review similar facilities to determine SSCs important to criticality safety, determine which parameters are most important through independent analyses.

Section x (page 33). Understand/review current guidance and precedence in this area. Understand technical issue with BWR burnup credit. Become familiar with validation for burnup credit and use of commercial reactor critical data. Perform preliminary scoping and sensitivity analyses in this area. Also staff should understand potential consequences of a criticality in post closure.

Section x (page 33). Identify what damaged fuel configurations have the potential to occur in during the preclosure and postclosure time periods as opposed to what is being modeled for transportation accidents.

Section x (page 34). Review material properties of neutron absorbers and their qualification for use in the repository environment. Also determine limitations of cross section libraries/benchmark experiments.

Section x (page 35). Understand the differences in criticality control methods, use of different calculational methods, and differences in validation methods for both criticality analyses as well as source term/shielding analyses.

Section x (page 36). Address in a systematic way whether any of the conditions above (and possibly others) have the potential to affect the release and transport of important radionuclides, in particular neptunium.

References:

1. Standard Review Plan (SRP) For Dry Cask Storage Systems, NUREG-1536, U.S. Nuclear Regulatory Commission, January 1997.

2. Spent Fuel Project Office Interim Staff Guidance - 8, Rev. 2 - Limited Burnup Credit, USNRC.

3. C.V. Parks, M.D. DeHart, and J.C. Wagner, Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel, NUREG/CR-6665, U.S. NRC, February 2000.

4. S.M. Bowman, W.C. Jordan, J.F. Mincey, C.V. Parks, L.M. Petrie, Experience with the SCALE Criticality Safety Cross-Section Libraries, NUREG/CR-6686, Oak Ridge October 2000

5. I.C. Gauld, J.C. Ryman, Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel, NUREG/CR-6700, Oak Ridge January 2001

6. Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses, NUREG/CR-6781, Oak Ridge January 2003

7. ASTM C 1671 – 07 Standard Practice forQualification and Acceptance of Boron Based MetallicNeutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging

8. Technical Work Plan for Development of Technical Data Needed to Justify Full Burnup Credit in Criticality Safety Licensing Analyses Involving Commerical Spent Nuclear Fuel, TWP-EBS-MD-000019 Rev 01, February 2007.