NRC/TN Meeting
CoC 9302 MP197HB
RAI Discussions
January 14, 2010

Transnuclear Inc.

AREVA

RAIs for Discussion (non-proprietary-information)

2-1
2-4
2-6, 2-22, 5-2, 7-2
2-7
2-8, 2-23
2-9, 8-2
2-10
2-11
2-19
2-20
2-21
2-22 (see 2-6)
2-23 (see 2-8)
2-27
2-28, 2-32
2-31
2-32 (see 2-28)
2-34
2-36
2-37

3-1
3-4
3-6
3-7
3-9, 3-12
3-10
3-11
3-12 (see 3-9)
3-15

4-1
4-2
4-3
4-4

5-2 (see 2-6)
5-3
5-4
5-5
5-6
5-7
5-8
5-9, 5-14
5-10
5-12
5-14 (see 5-9)
5-15
5-16
5-18
5-19
5-20

6-1
6-3

7-1
7-2 (see 2-6)
8-1
8-2 (see 2-9)

RAI 2-1 – Add additional information to the definition of damaged fuel

- TN's design incorporates 3 different types of fuel assemblies: failed, damaged and intact. See Appendix A.1.4.3 for the definition of failed fuel. Note that damaged fuel assemblies are stored with end caps and failed fuel assemblies are stored in the failed fuel cans
- The current definition of damaged fuel is consistent with that employed for storage in the CoC-1004. This definition was already expanded to include fuel assemblies containing the fuel hardware attached intact (presence of top and bottom nozzles)
- Current definition in the SAR: "assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is able to be handled by normal means"
- Fuel with damage greater than those indicated will not be considered for loading as damaged fuel assemblies
- Separate evaluations – Criticality, Shielding, Structural etc. are considered for damaged fuel assemblies

RAI 2-4 Neutron Absorber Testing

- A large number of DSCs addressed in this application are in use under Part 72 storage requirements which were fabricated in accordance with the requirements in force at the time of fabrication. The requirements for most of these DSCs agree with those given in CoC 1004, Amendment 10. So these requirements can be included in the MP197HB SAR.
- However, there are a few DSCs that were fabricated using slightly different neutron absorber acceptance testing requirements. So these Part 72 requirements specific to the particular DSCs will be included as exceptions to the above requirements.
- The two DSCs that are yet to be licensed (37PTH and 69BTH) are derivative designs and the neutron absorber test requirements noted in this application are the same as the originating DSCs.

RAI 2-6, 2-22, 5-2 and 7-2
Radioactive Waste Canister

- The Radioactive Waste Container (RWC), as described in Section A.1.2.3.2 is a secondary container.
- During transport, the only safety function provided by the RWC is additional shielding. SAR chapter A.5 will be modified to provide additional clarification of the shielding thicknesses employed in the models.
- The other design features of the RWC are governed by handling and temporary storage requirements (e.g., 10CFR50).
- RWC drawings will be added to the SAR to describe these features, including materials, appropriate fabrication details and codes of construction.
- A section describing galvanic interaction/gas generation will also be provided in Section A.2.2.2.
- A description of RWC loading and transport preparation processes will be added to Chapter A.7.

RAI 2-7

- The effect of the radial hydrides on the high burn fuel cladding mechanical properties are addressed in the CoC 1004, Amendment 10 RAI #1 (11/07/2007) (response to RAI 3-4) and RAI #2 (5/23/2008) (response to RAI 3-1)
- In the responses, it was concluded that:
  - the tensile strength in the axial direction does not change significantly as the percentage of the radial hydrides increases
  - there is a significant reduction, however, in tensile strength in the hoop direction and also ductility as the percentage of radial hydrides increases
  - using the Beyer's model for prediction of the irradiated fuel cladding mechanical properties, the yield strength in the hoop direction should be reduced by approximately 20%
  - the maximum calculated cladding hoop stresses for BWR and PWR fuel cladding due to internal pressure should be compared with the reduced BWR and PWR cladding yield strengths

RAI 2-7 Continued

- Proposed responses
  - The information in the responses to RAI #1 (RAI 3-4) and RAI #2 (RAI 3-1) will be added to Appendix A.2.13.11 (new Section will be added to this Appendix)
  - New tables will be added to this new Section:
    - Calculated maximum hoop stresses due to internal pressure for BWR and PWR fuel cladding
    - Compared this maximum hoop stress with the reduced BWR and PWR cladding yield strengths (20% reduction)
- Factor of safety will be calculated to show that the reduction in yield strength of the cladding due to the effect of radial hydrides can be accommodated with significant margin

RAI 2-8 and RAI 2-23

- Addendums 10f, 10g, and 10h are used as references for damaged fuel cladding structural evaluations in Appendix A.2.13.11, Section A.2.13.11.4 in revision 5 of the SAR

- Evaluation in Section A.2.13.11.4 is to show that “the fuel can be retrieved and handled by normal means” even after the transportation normal or accident conditions, which is not necessary or required under 10CFR Part 71.

- In response to NRC Response to Supplemental Information (RSI) questions, this section and associated references, tables, and figures have been deleted. TN SAR changed pages (revision 6) to incorporate these deletions were included in the RSI responses (6/22/2009).

- TN will update in Chapter A.6 criticality analysis of the damaged fuel using the most credible configuration of damaged fuel assemblies consistent with the damaged fuel condition of the package

RAI 2-9 and 8-2
DSC Post-Storage Condition and Material Properties

- As part of the Part 72 licensing process, TN has demonstrated that stored DSCs are able perform their design functions throughout the storage period.
    - Stainless steel properties do not change with time under the operating conditions of storage.
    - In response to staff questions during the 1004 CoC Amendment 10 review, concerns over aluminum creep in the DSCs were resolved.
    - Neutron poison material has undergone considerable testing to assure its continued performance throughout storage.
- Many of the design functions required for storage are also required for transport and since the DSCs remain functional at the end of storage, the ability to meet these requirements during transport is assured.
- The ability of the DSCs to meet requirements unique to transport have been demonstrated in this application.

RAI 2-9 and 8-2 (cont.)

- Removal of the DSCs from storage has been evaluated during the Part 72 licensing process. It was shown that DSCs are not damaged during insertion and removal from the storage modules.
- During insertion and extraction of the DSC from the HSM, the force is monitored to ensure that the DSC is not damaged.
- Additional data will be provided that demonstrate the ability of VYAL B to maintain its shielding ability over time.
- Operating procedures in Chapter A.7 are revised to add the requirement that for the DSCs which are in storage before transportation, evaluation is performed to validate that the DSC shell is within the analyzed condition.

RAI 2-10 Wood Properties

- Wood conductivity and its dependency on the grain direction and moisture content is discussed in SAR, Section 3.2, Item 8, Page 3-7 as noted in Section A.3.2.1, Item 14.
- The thermal diffusivity of wood is 0.00025 in^2/s (0.9 in^2/hr) based on "Wood Handbook", Page 3-17. This value will be added to Section A.3.2.1, Item 14.
- References for wood data will be added to Tables A.2.13.12-5 and A.2.13.12-6.

RAI 2-11 Conductivity Tables in Section A.3.2.1

- The properties listed in Section A.3.2.1 are itemized from 1 to 31. Each item is valid for all DSC types unless it is specifically noted in the title of the item. A note will be added at the introduction of this section to avoid any misinterpretation.

RAI 2-19 Reduce the burnup from 70 GWD/MTU to 62.5 GWD/MTU

- Isotopic assay benchmarks available at higher burnups
- The thermal analysis demonstrates margins in the range of 100 °F
- High burnup related damage to fuel has already been considered in the safety analyses – structural, shielding, criticality
- The staff has approved applications with maximum burnup greater than 62.5 GWD/MTU

RAI 2-20 Lead Pour and Resulting Gaps

- The lead pour and cool down processes are tightly controlled to ensure the lead cools from the bottom of the pour to the top at a controlled rate and that the advancing boundary of solidified lead is always covered with molten lead. The vertical standpipes above the lead cavity through which the molten lead flows are the last to cool, so that there is no gap left at the top of the lead cavity after the pour is finished.
- The volume between inner and outer shells is thus completely filled with lead at the moment the lead solidifies.
- All subsequent calculations use solid lead properties at appropriate temperatures.
- This procedure assures correct gap estimates for both thermal and slump evaluations.

RAI 2-21 Damaged Fuel End Caps

- The MP197HB transport package contents include DSCs that may contain either failed or damaged fuel, depending on the DSC types.
- Failed fuel is contained in a failed fuel can that can be lifted with its contents from the DSC.
- Damaged fuel is transported in specific fuel compartments of a DSC that are closed, top and bottom, with end caps. These caps are not welded to the fuel compartment. Instead, they are designed so that the axial gap between the DSC basket and DSC top and bottom ends is small enough that the end caps remain inserted in the fuel compartment, thus containing any loose material from the damaged fuel assembly.
- Additional drawings and descriptions will be added to clarify this.

Basket Analysis - RAI 2-27
Provide justification of high stresses for NCT

- The exceedance for the stresses are localized and have characteristics of secondary stresses
- Limit analysis (NG-3228.2) is used to justify these high stresses
- The explanation is provided in Section A.2.13.8.6.1.D
- A note is provided at the bottom of Table 2.13.8-12 which states “Limit Load Analysis is performed for these cases”

Impact Limiter Analysis

- RAI 2-28 and 2-32 – Explanation of the strain failure criteria
- RAI 2-31 – Visible Hourglassing in the 1/3 slapdown LS-DYNA analysis
- RAI 2-36 – Comparison between peak g-load from LS-DYNA vs average from all accelerometers
- RAI 2-37 – Node numbers and locations used to evaluate the accelerations

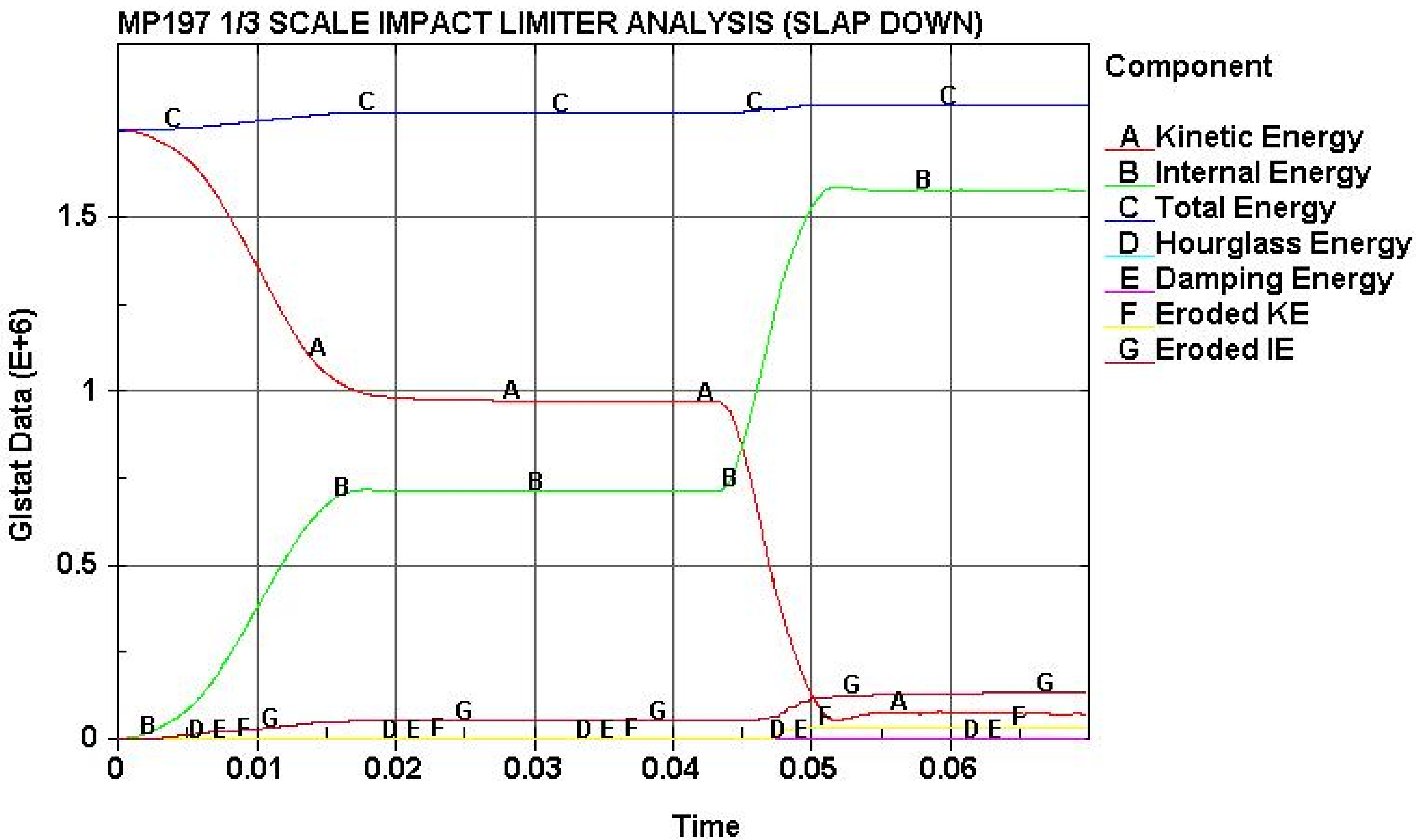
Impact Limiter Analysis - RAI 2-28 and 2-32
Tensile Failure Strain Option

- Only the tensile failure strain option is used for the wood materials
- Some form of element failure is needed to avoid excessive distortion of the mesh
- Failures are generally parallel to the wood grain, which is a realistic failure mode for the wood
- Sensitivity Analysis will be performed to study the affect of the strain limit values

Impact Limiter Analysis - RAI 2-31
Hourglass

- Selectively reduced elements were chosen to reduce hourglassing
- The energy results for the 1/3 slapdown case show that the element distortions are a result of contact issues rather than hourglassing
- Contact definitions will be modified so as to resolve distortion
- Sensitivity Analysis will be performed to study the effect of selectively reduced elements vs constant stress elements

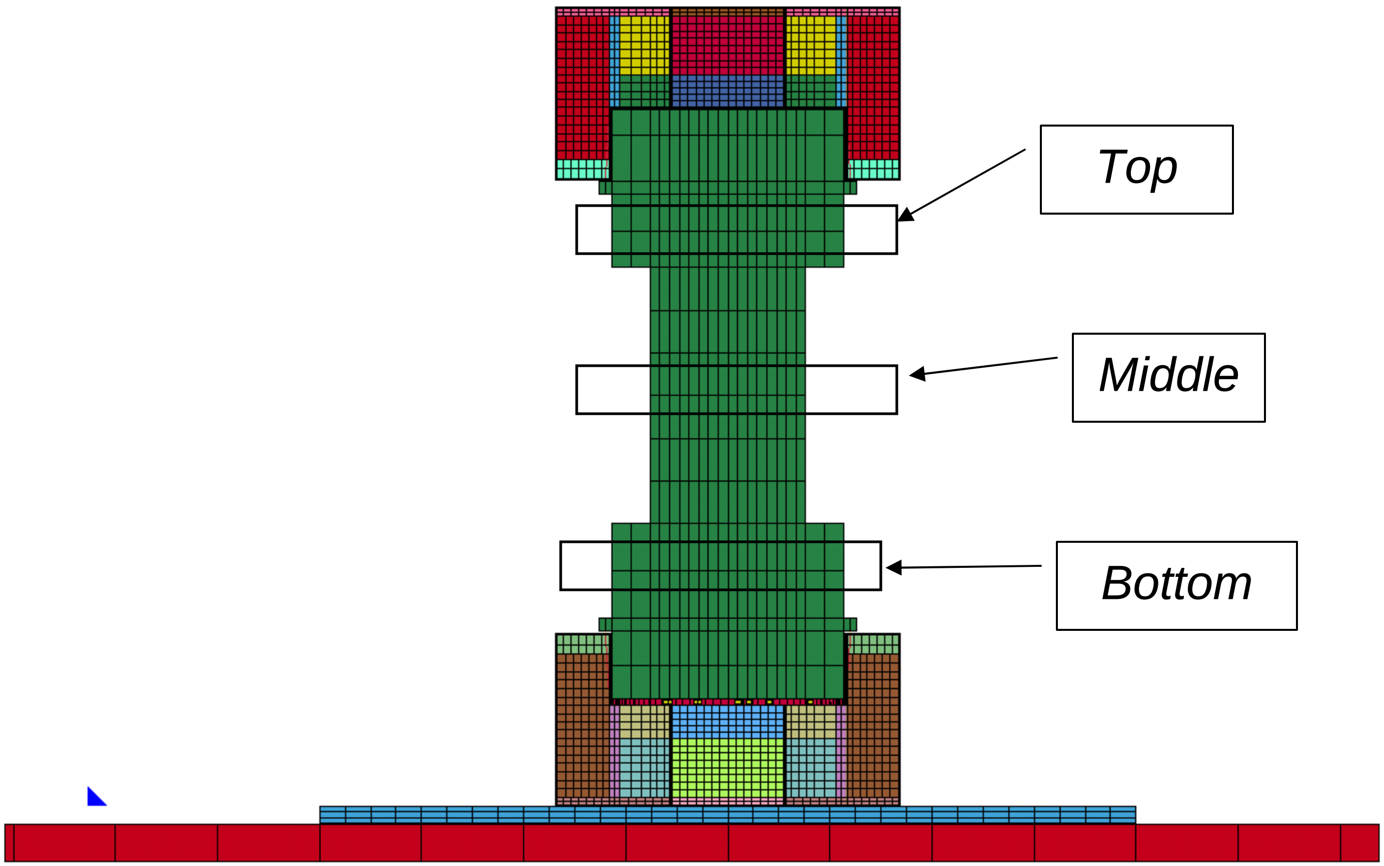
Impact Limiter Analysis - RAI 2-31 (Continued)
Hourglass



RAI 2-34 NUH61BT DSC Overall Length

- The approximate 199.7” dimension given in Section 1.4.7 agrees with the dimensions provided on drawing NUH61BT-71-003 of 196.04” max plus (3.75”) totaling 199.79”.

Impact Limiter Analysis - RAI 2-36 and 2-37
Node Locations for Accelerations



Impact Limiter Analysis - RAI 2-36 and 2-37
Node Locations for Accelerations

- The accelerations at all nodes in middle region were averaged for the end and side drop analysis and compared against the averaged accelerations from the accelerometer
- The accelerations at all of nodes in top and bottom regions were averaged for the slapdown analysis and compared against the averaged accelerations from the accelerometer at the top and bottom of the cask, respectively.
- The locations of the nodes for the top, middle, and bottom regions will be added to the SAR

RAI 3-1 Material Properties

- The conductivity value of 11.150 Btu/hr-in-°F is used only for 100 °F in the input files of the ANSYS models for the baskets. The other values for other temperatures are correctly applied in the input files. The intention of the note on page A.3-12 was to show that error in the input file for the aluminum conductivity at 100 °F does not have any effect on the results since all the component temperatures of the baskets are much higher than 100 °F. This is evident from temperature profiles shown in SAR Figures A.3-31, A.3-32, and A.3-34.
- Note 1 on page A.3-12 will be revised for clarification.

RAI 3-4 Hot Stream on Personnel Barrier

- The boundary layer thickness for the cask with the highest heat load will be calculated.
- This calculation will demonstrate that the boundary layer thickness is smaller than the distance between the cask and the personnel barrier.
- SAR Section A.3.3.1.2 will be revised to reflect the result of this calculation.

RAI 3-6 Thermal Evaluation of DSCs under NCT

- Among the DSC types previously evaluated for storage applications and proposed for transport in MP197HB, 24PTH Type 2 has the smallest margin (19 °F*)* for the maximum fuel cladding under storage conditions and has the highest heat load for transportation conditions (26 kW). A thermal analysis of this DSC type under NCT is prepared to provide additional assurance that the arguments and evaluations reported in the SAR are valid.

- The 24PTH DSC shell temperature profile under NCT calculated in MP197HB model is mapped onto the 3D DSC model of 24PTH, Type *2.* The DSC model is identical to the model previously used for storage conditions in 10 CFR 72 SAR.

- A uniform heat load zone configuration with the maximum heat load of 26 kW is applied in the DSC model.

- A comparison of the maximum DSC component temperatures is listed in the following table.

RAI 3-6 Thermal Evaluation of DSCs under NCT

DSC Type	24PTH, Type 2		
Operating Condition	Normal Transfer 31.2 kW	NCT 26 kW	
HLZC	Uniform (1.3 kW/FA)	Uniform (1.08 kW/FA)	
Max. Comp.Temp.	$T_{Transfer}$ (° F)	T_{NCT} (°F)	($T_{Transfer} - T_{NCT}$) (°F)
Fuel Cladding	733	658	+74
Fuel Compartment	682	610	+72
Al/Poison	681	609	+72
Basket Rail	576	489	+87
DSC Shell	477	463	+14

RAI 3-6 Thermal Evaluation of DSCs under NCT

- As seen in the table, the maximum fuel cladding and basket component temperatures for the 24PTH Type 2 DSC under NCT is more than 70 °F lower than the bounding values listed in the SAR Table A.3-10.
- The reduction of the heat load from storage application to transport application is larger for the other high heat load DSC types such as the 32PTH1 or 61BTH. Therefore, a larger margin is expected for these DSC types.
- This analysis provides further evidence that the SAR approach to use bounding values from transfer conditions for transport conditions is conservative, although the corresponding DSC shell temperature profiles during transfer and transport conditions are not identical.
- SAR Chapter A.3 will be revised to reflect the above analysis and results.

RAI 3-7 Sensitivity of the Gaps

- Axial and radial gaps are considered in the MP197HB model. The axial gaps are located toward the ends of the cask and the radial gaps are located between the multiple shells of the cask. The peak cladding temperature is more sensitive to the radial gaps since the cask ends are covered with impact limiters, which act as insulators.

- A thermal test is planned to assure that the thermal performance of the as-built cask satisfies or exceeds the predicted performance in the radial direction. The thermal test procedures will be similar to those used in the testing of TN-32 casks presented to the NRC in previous applications.

- SAR Chapter A.8 will be revised to address the planned thermal test.

- The gaps assumed in the DSC models are the same as those assumed for storage applications. The validity of these assumptions were reviewed by the NRC in 10 CFR 72 applications.

RAI 3-9 and 3-12 Decay heat profile and Coupled Model

- A coupled model of the HAC is prepared to include the fuel assemblies, basket, and basket rails within the cask model based on the bounding 69BTH DSC as suggested.

- The results of the coupled model in comparison to the uncoupled model show that the peak cladding temperature decreases by a few degrees and the seal temperatures increase by a few degrees. This is because the total amount of axial heat transfer from the DSC ends toward the cask ends is small and is less than 5% of the total heat load. All component temperatures remain within the allowable limits.

- The reason for the changes are due to the fact that the un-coupled model (with homogenized basket) does not consider any heat transfer from the basket toward the DSC end plates, while in the coupled model axial heat transfer is unrestricted.

- SAR Section A.3.4 will be appended to include the above sensitivity analysis and its comparison to the original model.

RAI 3-10 and 3-11 Basket Temperatures

- The values reported on page A.3-133 are the average temperatures of the compartments and wrap plates, not the maximum temperatures.
- All of the compartments are loaded with fuel assemblies in the 69BTH DSC with HLZC#1 (26 kW), while the 69BTH DSC with HLZC#4 (32 kW) contains only 52 assemblies, in which 17 of the core compartments remain unloaded (empty/dummy assemblies). The core unloaded compartments have lower temperatures than the outer compartments. This results in an average temperature of the basket plates for HLZC#4, which is coincidently the same as the average temperature of the basket plates for HLZC#1.
- Since the total heat load ultimately is rejected through the DSC shell, the average DSC shell temperature for HLZC#4 (32 kW) is much higher than the average DSC shell temperature for HLZC#1 (26 kW).

RAI 3-10 and 3-11 Basket Temperatures

- Having the same average basket temperature but higher DSC shell temperature for HLZ#4 (32 kW) in comparison to HLZC#1 (26 kW) results in slightly larger radial gap between the basket and the DSC shell for HLZC#4.
- The hot gap sizes calculated for 69BTH DSC on both ends of the loading spectrum (HLZC#4 with 32KW and HLZC#1 with 26 kW) are lower than the assumed hot gap of 0.3”. Therefore, the assumed hot gap is also bounding for higher heat loads.

RAI 3-15 Thermal Test

- A thermal test will be added to Chapter A.8. This test will assure that the thermal performance of the as-built cask satisfies or exceeds the performance predicted by the model.

RAI 4-1, 4-2, 4-3 and 4-4
Containment Boundary O-rings

- Fluorocarbon o-rings provide the best solution when temperature and permeability requirements are considered.
- Radiation resistance of fluorocarbon is adequate and for this application, the exposure is limited (estimated to be ~$1x10^5$ rad) due to short transport times and one-time usage of the o-rings. The exposure damage threshold is between $1x10^6$ rad and $1x10^7$ rad.
- Current o-ring callout on drawings will be expanded to include Parker Compound V0835-75 equivalent.
- Recommended temperature limits for VM835-75 are -40°F to 400°F (Parker Material Report KJ0835).
- Additional clarification will be added to Chapter A.4, including appropriate temperatures with justifications.

RAI 5-3: DB Fuel and DB DSC

- Section A.5-2 of the SAR employs the MTU loading as an index of comparison for selection of the design basis fuel assembly
- The design basis DSC is based on the radiation source terms and shielding design only
- The PWR DSCs (where burnup is credited for criticality) require at least 15 years cooled fuel
- The allowable heat load for the 69BTH is the highest
- The cooling time requirements for the 69BTH are the lowest

RAI 5-4, RAI 5-5 and RAI 5-6

- Lead Slump considered in the shielding evaluation is more conservative that what is determined in the structural analysis – no tests needed
- Wood crushing considered in the shielding evaluation is more conservative that what is determined in the structural analysis – no tests needed
- Loss of neutron shielding considered is also more conservative than what is supported by fire test evaluation results
    - fire tests on the neutron shielding material indicate that fire does not propagate (the resin is a fire extinguisher) and that there is only a local effect- charring followed by de-gassing
    - structural evaluation demonstrates that the neutron shield shell remains intact
    - In general, the neutron shielding does not "collapse" – only 25% is conservatively credited in the HAC shielding evaluations

RAI 5-7: Differences in NCT and HAC Source Terms

- Under NCT, the neutron and gamma shielding of the cask remains unchanged
- Under HAC, a slight degradation in gamma shielding and substantial degradation in neutron shielding is employed
- HAC evaluations need to be performed where the contribution from neutron sources is maximized
- Fuel Qualification for the various DSCs is based on “approximately” the same dose rate at NCT
- Therefore, a different set of source terms that contribute to 95% of neutron dose rate at NCT is chosen for HAC

RAI 5-8 Source Term Uncertainties with SAS2H

- Uncertainty associated with isotopic concentration is only one method for source term validation
- The 10% uncertainty indicates the accuracy of the SAS2H code for calculating source terms
- Conservatism is included in the SAS2H models that serves to overcome these uncertainties
- Typical measurements with loaded casks always have yielded substantially lower dose rates

RAI 5-9, RAI 5-14

- The axial peaking factors for PWR fuel assemblies are based on axial burnup profile database. This includes fuel assemblies from a variety of US reactors that also contain axial blankets. This burnup profile is considered to be applicable to all PWRs

- The BWR burnup profiles are also bounding and are based on low-medium burnup. At higher burnups, the shape is expected to be benign and less bounding

- For gamma, the axial flux distribution is assumed to be proportional to the burnup. For neutrons, the axial flux distribution is assumed to be proportional to the fourth power of burnup

- The axial peaking factors in MCNP are applied as a normalized relative flux distribution in the active fuel region (RAI 5-14)

RAI 5-10: Energy Dependence of Source Terms

- The NCT and HAC source term calculations on Page A.5-9 are to determine the total source strength
- This is a scalar multiplier and is employed to calculate the dose rates
- The spectrum is provided in Table A.5-11 and Table A.5-12 for NCT and HAC

RAI 5-12 Soluble Boron in BWR Depletion

- Soluble boron is employed in BWR depletion to account for the presence of control blades between BWR bundles
- The presence of significant control rod movement throughout the BWR depletion cycle makes it necessary to include adequate treatment
- Using Soluble Boron to account for control rod movement is both simple and accurate
- This methodology has been employed in the Part 72 source term calculations for BWRs

RAI 5-15 Neglecting Voids in Homogenization

- Fuel assembly homogenization is performed to simplify modeling without affecting the dose rate results
- The discussion in Page A.5-13 on voids will be clarified
- The intent was to neglect the mass associated with these gaps and include their volume
- In summary, the homogenization is performed using the correct values for the component material densities

RAI 5-16 Evaluation for Plastic Deformation

- The structural analysis results will demonstrate that the fuel assembly will retain its integrity under NCT and HAC
- Local plastic deformation of the fuel rods does not have any impact on the axial source distribution
- The shielding evaluation will be revised to consider axial changes to source distribution under HAC for damaged assemblies

RAI 5-18 Response Function

- A new Appendix will be added to Chapter A.5 to describe the Response Function Methodology
- Response Function is simply a dose rate ranking methodology and is based on MCNP results as a function of DSC Type and Energy Group
- Fuel Qualification Tables (FQTs) for shielding are generated using the Response Functions for each DSC to determine acceptable combinations of Burnup Enrichment and Cooling Time (BECT)
- The FQTs are designed such that any given BECT combination results in approximately the same dose rate at 2m under NCT
- The dose rates from Response Functions will be presented in the Appendix including the neutron and gamma contributions
- The dose rate calculated from Response Functions will also be compared to the MCNP results for selected BECT combinations

RAI 5-18 Response Function

Source Geometry (Ex: Multi-Zone)

Source Spectrum (Neutron, Gamma)

MCNP Models for each DSC Payload within the MP197HB Cask

MCNP Evaluation of the MP197HB Cask for each DSC Payload

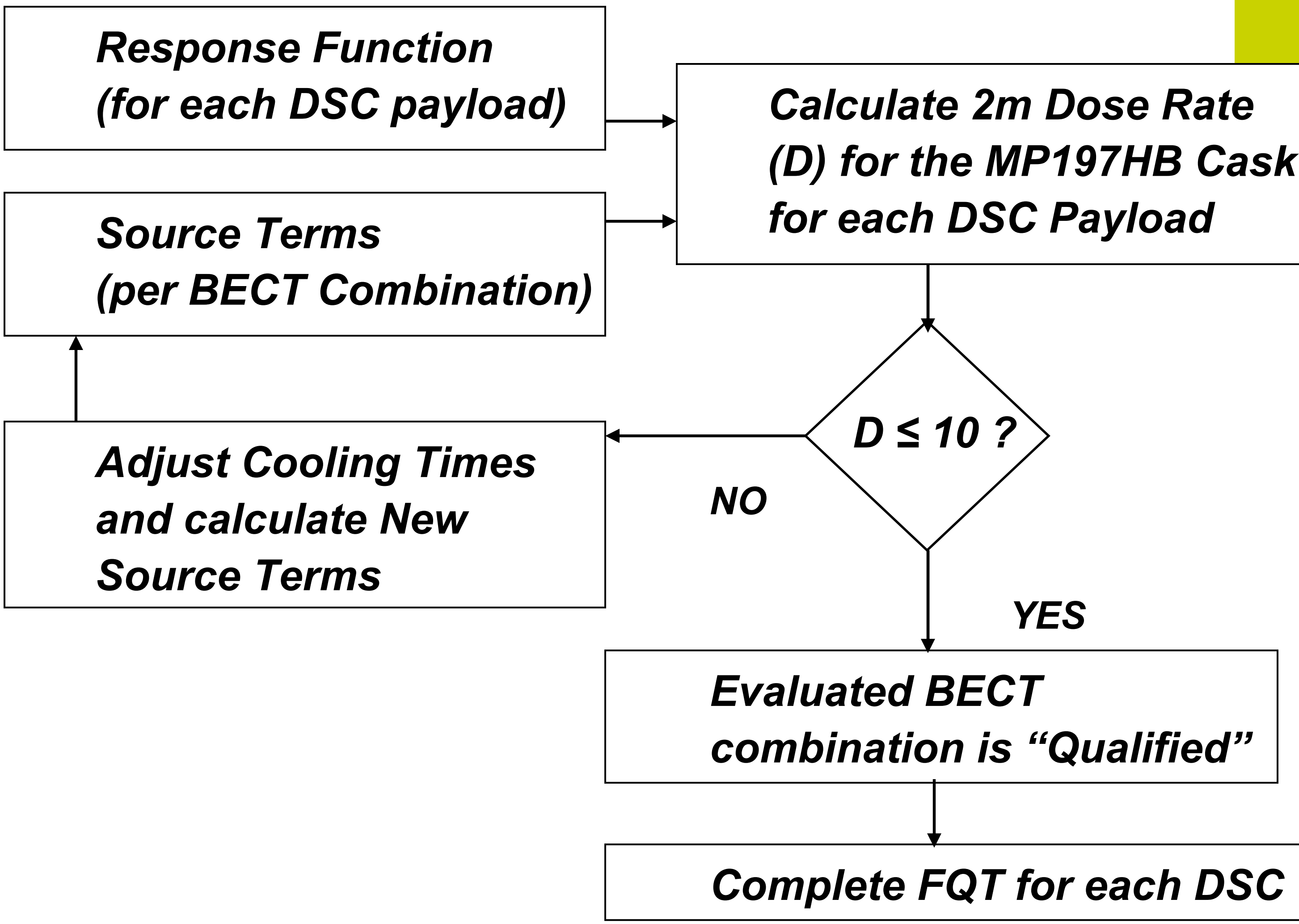
Response Function at 2m calculated for each DSC payload

RAI 5-18 Response Function

- ***An Example 4-Zone Response Function (2 m radial dose point) for the 69BTH DSC is shown in the Table below***

Zone	Neutron	Capture Gamma	Primary Gamma Energy Range (MeV)			
			1.00 to 1.33	1.33 to 1.66	2.00 to 2.50	2.50 to 3.00
1	4.65E-10	4.88E-10	6.05E-19	4.72E-18	1.45E-16	5.75E-16
2	1.61E-09	1.20E-09	5.18E-17	1.71E-16	5.5E-15	1.69E-14
3	4.17E-09	2.22E-09	2.95E-16	5.12E-15	8.01E-14	1.77E-13
4	1.04E-08	3.61E-09	9.89E-15	8.92E-14	8.14E-13	1.64E-12

RAI 5-18 Response Function



RAI 5-19 Axial Blankets

- For a given burnup, the presence of axial blankets will result in higher source terms since the burnup in the non-blanket regions will be higher
- For a given enrichment, the presence of axial blankets will result in lower source terms since the enrichment in the non-blanket regions will be higher
- The FQT (the basis for selecting fuel assemblies eligible for loading) requires that the enrichment is based on minimum bundle average and partially offsets the burnup distribution changes
- Sensitivity calculations will be performed to determine the additional changes to source terms (if any) that may be needed for fuel assemblies containing blankets

RAI 5-20 Decay Heat Equation

- The Decay Heat Equation is based on a numerical fit of numerous SAS2H results
- The calculation of the Decay Heat Equation contains tabulated data that shows the burnup, enrichment, cooling time, decay heat from SAS2H, decay heat from equation and the difference
- This calculation including data will be provided as part of the response

RAI 6-1: Transportation of Fuel with burnups > 45 GWD/MTU, ISG-19

- The application will be revised to include the moderator exclusion provisions under HAC given in ISG-19
  - The MP-197HB cask containment boundary is employed as the basis for the double containment requirement
  - In addition, the DSCs are welded systems and are tested to be leak tight
  - The cask structural analysis demonstrates that the cask containment boundary does not experience any leakage following HAC
  - The basket structural analysis demonstrates that the basket retains its geometry following HAC
  - A criticality analysis will be performed with representative fuel reconfiguration following moderator in-leakage (HAC) for defense-in-depth

RAI 6-3 Listing of the Limiting Parameters

- All fuel assembly (class) design information considered for loading are included in Chapter A.1
- The design details are consistent with the details included for the Part 72 Technical Specifications
- The Safety analyses have demonstrated that the Fuel Rod design information is less significant – criticality analysis include sensitivity analyses
- The inclusion of rod design information is at a level of detail that does not significantly affect the analyses

RAI 7-1 DSC Loading Procedures

- Each DSC authorized for transport in the MP197HB will have a complete loading section provided in Chapter A.7.
- The loading procedures will be taken from the existing Part 72 CoC or, in the case of the 37PTH and 69BTH, will be consistent with other previously licensed DSCs.

RAI 8-1
Verification of Neutron Shield Fabrication

- As stated in the SAR, the adequacy of the neutron shield is largely based on process control. This control includes
  - Batch measurements (chemical composition and density)
  - Personnel qualification
  - Periodic process qualification (includes sectioning sample pours to verify consistency of the solidified material throughout the tube
- Additional data, testing results and process details will be provided to demonstrate that the neutron shielding resin as modeled in the shielding analyses is representative of the installed material, thus insuring adequate shielding capabilities of the MP197HB packaging.
- Note that the VYAL B material and installation processes are nearly identical to those of shielding material used successfully for other existing TN packagings.