From: "BONDRE Jayant" <jayant.bondre@transnuclear.com>
To: "Meraj Rahimi" <MXR2@nrc.gov>
Date: 4/3/06 5:37PM
Subject: RE: Meeting on TN-40

Mr. Rahimi,

Attached file contains our preliminary version of the proprietary information we are planning to discuss with the staff in the meeting. Please let me know if you have any questions.

Regards

Jayant Bondre
Transnuclear, Inc.
410-910-6881

-----Original Message-----
From: Meraj Rahimi [mailto:MXR2@nrc.gov]
Sent: Monday, April 03, 2006 7:22 AM
To: BONDRE Jayant
Cc: Robert Nelson
Subject: RE: Meeting on TN-40

Mr. Bondre,

I need to receive at least a preliminary version of the proprietary information that you're planning to discuss during the closed portion of the meeting. The staff needs to have a basis for closing the proprietary portions of the meeting to the public.

thanks,

Meraj Rahimi
Senior Project Manager
U.S. Nuclear Regulatory Commission
Spent Fuel Project Office
Spent Fuel Licensing Section
Tel. 301-415-2947
Fax 301-415-8555
Email MXR2@NRC.GOV

CC: "Robert Nelson" <RAN@nrc.gov>

Mail Envelope Properties (44319598.125 : 1 : 12581)

Subject:	RE: Meeting on TN-40
Creation Date:	4/3/06 5:36PM
From:	"BONDRE Jayant" <jayant.bondre@transnuclear.com>
Created By:	jayant.bondre@transnuclear.com

Recipients
nrc.gov
twf4_po.TWFN_DO
MXR2 (Meraj Rahimi)

nrc.gov
TWGWPO01.HQGWDO01
RAN CC (Robert Nelson)

Post Office	**Route**
twf4_po.TWFN_DO	nrc.gov
TWGWPO01.HQGWDO01	nrc.gov

Files	**Size**	**Date & Time**
MESSAGE	884	04/03/06 05:36PM
TN40-prelimiary proprietary NRC mtg.pdf		175768
Mime.822	243624	

Options

Expiration Date:	None
Priority:	Standard
Reply Requested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

TN-40 Burnup Credit Preliminary Criticality Analysis Summary

A burnup credit criticality analysis has been performed to qualify the Westinghouse 14x14 (WE14) class fuel assemblies for transportation in the TN-40 Cask.

The significant challenges of this burnup credit analysis as applicable to spent fuel transportation are provided below:

- Credit for the presence of a limited number fission products though the ISG recommends actinide-only burnup credit.
- Limited availability of adequate benchmarks which reduce the uncertainty in the use of isotopic number densities that describe the burned fuel assembly (radiochemical assay data).
- Limited availability of adequate benchmarks that reduce the uncertainty in the reactivity worth of each isotope present in the burned fuel assembly (validation of cross sections).
- Limited availability of adequate "burnt" criticality benchmarks.

The salient features of the criticality analysis that are aimed at addressing these challenges are summarized below:

- Conservative treatment for depletion calculations combined with the use of results from available radiochemical assay data.
- Reliance on reactor critical benchmark experiments that demonstrate the conservatisms inherent in the cross section methodology.
- The final k_{eff} is calculated to be subcritical with a significant margin.
- Additional margins to the final k_{eff} are listed, some of which are also calculated explicitly.

A summary of the computer codes and benchmarking is provided below:

- SCALE 4.4 computer code package has been utilized to perform the burnup credit criticality analysis.
- Fuel assembly depletion calculations performed using the SAS2H module and the 44-group ENDF-V cross section library.
- Isotopic number densities from SAS2H are scaled utilizing the "Best Estimate Correction Factors" provided in ORNL/TM-13317.
- Benchmark calculations for depletion methodology involved re-running of all the available SAS2H models utilized in ORNL/TM-12667, ORNL/TM-13317 and ORNL/TM-2001/259.
- A total of 26 isotopes have been utilized to describe a burned fuel assembly – 12 actinides, 13 fission products and oxygen.
- Cask criticality calculations performed using the CSAS25 module (KENO V.a) and the 44-group ENDF-V cross section library.
- The model includes a full three-dimensional representation of the fuel assembly within the basket and an infinite radial array of casks.
- A set of 142 UO_2 and MOX critical experiments to determine the USL. The results of ORNL/TM-12294 are incorporated by reference. This reference demonstrates that the SCALE computer code results in conservative estimation of k_{eff} from reactor critical experiments.

A summary of the depletion analysis calculational methodology and models is provided below:

- SAS2H is utilized to perform the depletion calculations. Path B modeling option is utilized.
- Axial burnup profiles from NUREG/CR-6801 are utilized to model the fuel assembly with varying axial burnup. The 18-zone models in the stated reference are reduced to 8 zones that have identical end zones and a flat-burnup central zone. This is done to reduce the number of zones without affecting the results.
- All the axial zones are modeled with a Specific power of 25 MW/Assembly, Fuel temperature of 901 K, Clad temperature of 620 K, Moderator temperature of 570 K, Moderator density of 0.733 g/cm^3 and a soluble boron concentration of 600 ppm.
- All depletion calculations were carried out for the WE14 Standard fuel assembly with BPRAs (containing 16 rods per BPRA, maximum). These BPRAs are assumed to be present in 2 out of the 3 depletion cycles.
- The reactivity effects of IFBA, when present, during depletion are conservatively ignored.
- The isotopic number densities obtained from SAS2H are scaled by the "best estimate factors" to yield the burned fuel isotopic number densities for use in the criticality analysis.
- Further, these number densities are conservatively adjusted by 1) rounding the best estimate correction factors for fissile isotopes (U-235, Pu-239 and Pu-241) up, to the next highest second decimal place or set to 1.0 which ever is greater; 2) rounding the best estimate correction factors for the absorber isotopes (when greater than 1.0) down, to 1.0; and 3) conservatively truncating all the factors to two decimal places from the best estimate correction factors.

A summary of the criticality analysis calculational methodology and models is provided below:

- CSAS25 is utilized to perform the criticality calculations.
- The TN-40 cask is modeled in full 3-D using the available geometry options.
- A 75% credit is taken for the presence of fixed neutron absorber in the basket.
- The most reactive geometry configuration is based on the fuel compartment with the minimum internal dimension (8.00″ with a thickness of 0.09″), the fuel assemblies in the "inward" position thereby minimizing their separation.
- An infinite array of casks is modeled in the radial direction via specular radial boundary conditions with water axial boundary conditions.
- Horizontal burnup "bias" is added to the final k_{eff} using a burnup dependent horizontal gradient. The Δk_{eff} due to horizontal burnup is conservatively determined using uniform burnup thereby maximizing the "bias".
- All calculations are performed with the design basis WE14 Standard fuel assembly within the TN-40 basket.
- Normal and Accident conditions are modeled assuming that the TN-40 and its contents (basket and fuel) are fully flooded in unborated water.
- Hypothetical accident conditions are based on no change to the cask and basket geometry except that the cask neutron shield is lost and the casks are pushed closer together.

- Optimum moderator density calculations are carried out with internal and external moderator density variations.
- The worst case k_{eff} value is 0.9356 and resulted from the 3.85 wt. %, 31 GWD/MTU case with an internal moderator density of 100% and an external moderator density of 90% of full density un-borated water.

The preliminary results of the analysis are as follows:

- Cask contents – WE14 class fuel assemblies with and without control components.
- Maximum enrichment of fuel assemblies is 3.85 wt. % U-235.
- Maximum burnup credited in the analysis is 31 GWD/MTU (this includes a conservative 5% burnup uncertainty).
- Minimum cooling time required is 15 years.
- Four burnup-enrichment combinations are explicitly analyzed – 1.65 wt. % U-235 at zero burnup (fresh fuel assumption), 2.30 wt. % U-235 at 16 GWD/MTU, 3.30 wt. % U-235 at 26 GWD/MTU and 3.85 wt. % U-235 at 31 GWD/MTU. These four combinations are fitted to a fourth order polynomial that determines the required burnup as a function of initial enrichment.
- The maximum k_{eff} including all applicable biases and uncertainties is 0.9356 and is below the USL value of 0.9415 and also includes a 5% margin (0.05 Δk_{eff}) for subcriticality.

A summary of the additional reactivity margins in the criticality analysis is provided below:

- An increase in the cooling time by 5 years (from 15 years to 20 years) shows that the reduction in k_{eff} is about 0.05.
- A comparison of the spent fuel loaded in the TN-40 casks indicates that on an average there are less than 5 fuel assemblies per cask that are loaded with the design basis parameters. The average burnups are at least 3 GWD/MTU greater than the design basis burnups analyzed. This results in a significant margin to the k_{eff}.
- All calculations are performed assuming the WE14 standard fuel assembly is loaded in all of the 40 assembly locations. An examination of the spent fuel inventory in the loaded casks indicates that the fuel assembly distribution is mixed and contains a significant number of the other WE14 fuel assembly designs (OFA, Exxon Standard, Exxon Top Rod etc.,) whose reactivities are lower than that of the WE14 standard assembly design.
- The depletion for burnup credit is carried out assuming 16 rods per BPRA are present for at least two out of three depletion cycles. In reality, the average number of rods per BPRA per fuel assembly is much lower thereby resulting in an increased margin to the calculated k_{eff}.
- No credit is taken for the presence of natural uranium or low enriched uranium blankets. The presence of blankets in the most reactive portion (top 6 to 8 inches) of the fuel assembly results in an increased margin to the calculated k_{eff}.
- All calculations were performed assuming that the fuel assemblies are loaded into the cask with no inserts. The presence of inserts like BPRAs results in a reduced moderation within the fuel lattice and consequently a reduction in the calculated k_{eff}.