Novembre 29, 2010

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10, route du panorama
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SUBJECT: COMMUNICATION WITH FRENCH SAFETY AUTHORITY (ASN) CONCERNING REGULATORY REVIEWS

Dear Ms. Clemente:

Per the request from Ms. Marie Thérèse Lizot, dated October 7, 2010, I am providing you with our responses to your questions regarding NRC reviews of radioactive material transportation packages. Please note that the information provided in the enclosure is a general synopsis of staff review methodologies. Please let me know if we can be of any further assistance. Please feel free to contact me at anytime at (301) 492-3323 or by email at kevin.witt@nrc.gov.

Sincerely,

/RA/

Kevin M. Witt
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards

Enclosure : As Stated

cc: Rick Boyle, Department of Transportation

November 29, 2010

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***see previous concurrence**

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Comparison of rules and recommendations on transports of radioactive material applicable to the United States and France

List of questions to the NRC

1. Generality

    a. Increase of ambient pressure: what is the reason for this requirement and the associated value (140 kPa)?

    NRC Response:
    10 CFR Part 71.71(c)(4), "Increased External Pressure," provides that an external pressure of 140 kPa absolute be considered for the evaluation of each package design under normal condition of transport (NCT).

    The requirement is intended for demonstrating that the package containment boundary will neither breach nor buckle when minimum internal pressure with ambient temperature of 29°C, zero decay heat, and zero insolation are considered together to result in the greatest pressure difference between the inside and outside of the containment system.

    Although the pressure increase of 140 kPa absolute has been in place since the March 1989 issuance of Regulatory Guide 7.8, Revision 1, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material."

2. Thermal analysis

    a. Thermal test – Initial conditions: the NUREG-1886 specifies the absence of insulation, but not the 10 CFR 71.73 (b). To be confirmed.

    NRC Response:
    The absence of insulation is specified in NUREG-1886 as a result of the Canadian regulations. The cited regulation does not require consideration of insulation. NUREG-1886 is the Joint Canada - United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages and is one acceptable way of meeting the regulations.

    b. Post-combustion of wood of shock absorbers (= combustion of wood after extinction of regulatory fire): is this phenomenon taken into account in the thermal analysis?

    NRC Response:
    Yes, 10 CFR 71.73(c)(4) requires that any combustion of materials be allowed to proceed until it terminates naturally.

    c. Confined transport (under tarpaulins or canopies): are there specific assessments for those more penalizing configurations in the U.S.?

    NRC Response:
    The NRC position is that controlling or limiting conditions of transport should be accurately modelled. However, current regulations do not require to applicant to describe the type of transport vehicle which will be used as a part of the application. The applicant should confirm that the use of an enclosure does not exceed the conditions for which their package has been evaluated prior to shipment.

Enclosure

3. Criticality analysis

   a. Enhanced water immersion test: according to the regulation, the in-leakage of water (penetration of water inside the cavity of the package) is not acceptable in the U.S., but due to the leakage rate of the package, the water a limited quantity of water will penetrate. In France, the mass of water leaking into during the immersion test is calculated and used as input data for the study of criticality. What about in the U.S.?

   NRC Response:
   10 CFR 71.55(b) requires that single undamaged fissile material transportation packages be evaluated assuming water in-leakage to the most reactive credible extent, regardless of the condition of the package under normal or hypothetical accident conditions. §71.55(d) and §71.55(e) requires that single fissile material transportation packages subjected to normal hypothetical accident conditions, respectively, remain subcritical assuming that water moderation occurs to the most reactive credible extent. §71.59 requires that arrays of packages subjected to both normal and hypothetical accident conditions of transport are demonstrated to be subcritical in determining the CSI.

   Applicants typically assume that water in-leakage does not occur under normal conditions for the single package evaluation of §71.55(d) and the normal conditions package array evaluations in §71.59, as it is typically easy to demonstrate that such packages do not leak under the fissile material package immersion requirements. Rather than subject damaged packages to the immersion test to demonstrate that they do not leak, applicants typically assume that any amount of water can leak into a damaged package for the single package evaluation of §71.55(e) and the damaged package array evaluation of §71.59. NRC staff consider it difficult to conclude that only a certain amount of water will leak into a damaged package, and have rejected such arguments in the past, thereby requiring applicants to evaluate both damaged single packages and package arrays with full water in-leakage.

   One exception to this practice is for spent nuclear fuel transportation packages, which are typically structurally robust and may be capable of remaining water-tight after an accident. NRC has allowed applicants to consider damaged spent fuel packages without water in-leakage for the single package and array evaluations, provided they meet criteria given in Interim Staff Guidance 19, Moderator Exclusion Under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e) (http://www.nrc.gov/reading-rm/doc-collections/isg/isg-19.pdf).

   b. Is the Burn-Up Credit taken into account and under which hypotheses or conditions?

   NRC Response:
   NRC's Interim Staff Guidance (ISG) #8, Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks (http://www.nrc.gov/reading-rm/doc-collections/isg/isg-8R2.pdf), provides criteria that are acceptable to the staff for "actinide-only" burnup credit in PWR spent fuel transportation packages. This guidance limits burnup credit to eight major actinides (235U, 238U, 238Pu, 239Pu, 240Pu, 241Pu, 242Pu, and 241Am), 5.0 weight percent initial enrichment fuel, 50 GWd/MTU burnup, and 40 years of cooling time. This guidance also recommends a measurement prior to loading to confirm the reactor record burnup.

   Staff have approved two burnup credit package designs, and are in various stages of review for three others. None of these package designs followed all of the major criteria in ISG-8, but instead provided justification for going beyond this guidance (e.g., crediting fission products, performing a misload analysis in lieu of a measurement). Additionally, while the lack of available isotopic depletion and criticality code validation data caused staff to *recommend actinide-only credit in the current version of ISG-8, more such data has come available in recent years, allowing staff to consider revising the ISG to allow for fission product credit. To handle the continuing lack of applicable critical experiments containing spent fuel fission product concentrations, staff has contracted Oak Ridge National Laboratory to develop a criticality validation methodology based on using sensitivity and uncertainty analysis tools to propagate nuclear data uncertainties into an estimate of fission product criticality code bias and bias uncertainty. Staff expects to include this methodology in revised guidance by fall of next year.*

4. Mechanical analysis

   a. Conformity and approval of the targets for drop tests: what is the current practice in the U.S.? In France, assessment of "shape factor" and requirements for justification of relative hardness.

NRC Response:
On verifying test procedures, test equipment and the impact pad, Section 2.5.4.2 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Fuel," notes that the 1995 Lawrence Livermore National Laboratories (LLNL) Report, UCRL-ID-121673, provides guidelines for package drop testing. Section 5.6, "Unyielding Target and Puncture Bar," of the LLNL report provides clarification for an unyielding target for free drop tests of a shipping package, making reference also to the IAEA Safety Series No. 37 guidance.
The NRC staff is aware of the target mass and surface hardness properties, which meet the IAEA Safety Series No 37 guidance, for most of the testing facilities used by the shipping package designers. As a routine practice in reviewing target adequacy for packaging free drop tests, the staff continues to follow the guidance provided in the LLLN report and IAEA Safety Series No. 37.

   b. Scale models: according to the NUREG-1886, it must be proven that the tests will give conservative results. What are the important parameters? (scaling factor on dimensions, conservation of total drop energy taking into account masses and penetration/crushing heights, …)

NRC Response:
Because of the large variety of packages and the many different approaches that can be taken to evaluate these packages, no single set of "important parameters" can address in detail every situation that might be applicable to scale model drop tests. Appendices A1 through A8 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," identify package safety functions and typical areas of safety review for eight package types. These appendices have been used by the staff in determining major safety features unique to a package, including those needed to be considered in the free drop tests.
The LLNL report cited in responding to Question 4.a above provides comprehensive guidance, in Appendix B, "Scale-Model Impact Testing," for conducting impact testing on shipping packages. The staff safety evaluation considers this guidance to ensure that important free drop parameters are conservatively implemented in the tests.

   c. Behaviour of shock absorbers in temperature: in France, we consider a correction factor of the crushing stress of wood as a function of the temperature (for example, for balsa wood, 0.85 at 80°C and 1.4 at -40°C). Are similar benchmark values currently used in the U.S.?

NRC Response:
Yes, as per the wood handbook
If prevailing ambient temperature for the drop test was different from the heat (38°C) and temperature cold (-29°C) conditions, the impact limiter material crush strength would be adjusted in an analytical extrapolation for estimating prototypical response of a package.

Crush strength temperature effects are based on testing sample wood cubes. In the U.S., if the impact limiter was not chilled to an ambient temperature of -29^{o}C, a wood crush strength adjustment by +15% would commonly be considered for estimating the rigid body deceleration response of the package for the temperature cold of -29^{o}C. For the heat condition, a wood strength reduction by 15% would be used in the estimate to ensure that the impact limiter would not exhaust its deformation capability of protecting the packaging from potential hard landing onto the target.

d. Analysis of the impact of transport frame and lifting and tie-down devices on the safety of the packages: Are there specific analysis? In France, it should be demonstrated that the transport frame have no significant impact on the mechanical behaviour of the package during drop test.

NRC Response:
No. They are not specific analysis. 10 CFR 71.73(c)(1) provides that the HAC free drop tests must be conducted through a distance of 9 m onto a flat, essentially unyielding, horizontal surface in a position for which maximum damage is expected. Depending on package types, structural performance effects of package appendages, such as transport frames and lifting and tie-down devices, which are a structural part of the package, are routinely considered in the staff safety review.
For prototypical packaging demonstration by tests, the staff review bases for the applicant's evaluation of the most damaging drop orientations for selecting a limited number of bounding test orientations to ensure that packaging features most susceptible to free drops are challenged. This would involve, for instance, considering the presence of support skids and fork lift channels for unirradiated fuel, closure ring bolt and lockwire for uranium oxide drum, and integral base plate and box lifting frame for special form packages.
For large spent fuel casks with fixed lifting trunnions, cask vendors would provide sufficient recesses in the inner side of an impact limiter to ensure that trunnions will not cause the impact limiter to lock up, thereby, sharply raising the deceleration response of the cask, upon landing onto the unyielding target. Those special design features have been configured for scale-models used in drop tests for determining rigid body cask deceleration responses. They are also considered in analyzing cask structural performance for which cask side and slap-down drops are evaluated.

5. Absence of leakage of packages/ Activity release

a. is there any "relaxed criterion" accepted in the U.S.: for example evaluation of activity release on the basis of the leakage rate controlled in maintenance instead of the one controlled before expedition or no activity evaluation in the SAR if the leakage rate controlled before expedition is sufficiently low?

NRC Response:
No activity release assumed for packages that are classified as leaktight per ANSI N14.5.

b. Gasket extrusion, Expansion coefficients of gaskets: are there tabulated values in the U.S.?

NRC Response:
Most applicants rely on manufacturers' specifications, but the NRC may also request an applicant to perform testing to qualify gasket use at higher or lower temperatures, that are outside of normal temperature ranges of operation.

c. Behaviour of gaskets in temperature: are there specific studies to qualify the behaviour of the various grades of elastomer gasket in temperatures (high or low temperature)? Is there an evaluation of the allowable lifetime of gasket depending on the temperature reached in normal conditions of transport?

NRC Response:
BAM (Germany) has done some significant work in this area. They recently had an article in the International Journal for Packaging, Transport, Storage, and Security of Radioactive Material.

To date manufacturers specifications have been accepted for normal temperature range of operation. Testing is being done to confirm this for temperatures beyond regulatory concerns. It is also a question for the NRC project to conduct further research into extended storage and transportation where the issue is being evaluated for the gap analysis.

d. Compression rate of gaskets: what are the benchmark values in the U.S.?

NRC Response:
*On demonstrating adequate structural performance to satisfy containment requirements of 10 CFR Part 71, Section 2.5.7 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," provides that, "Inelastic deformation of the containment closure and seal system is **generally unacceptable** …" (boldface added). The NRC staff review experience indicates, however, that certain packages, such as medical isotope casks with integral overpack impact protection capability, may be subject to high deceleration inertia force challenges during the hypothetical accident condition (HAC) free drops. This generally will result in small separation at the mating surface between the container flange and the closure lid. To make safety findings on acceptable separation or gap opening, the NRC staff evaluates the O-ring seal compression performance for having sufficient elasticity to continue to fill the calculated gap associated with the HAC cask free drops.*
For the separation calculated by a quasi-static finite element analysis (FEA), the gap opening is considered acceptable if it is sufficiently less than the difference between the original design deflection and the "compression set" of the O-ring seal. As compression sets and many other seal performance features are product and operating conditions specific, there are no benchmark compression set values established to guide staff review. Rather, the staff would make safety findings on seal performance based on engineering data in applicant cited manufacturers' catalogues, including "Parker O-ring Handbook, ORD 5700/USA" of Parker Hannifin Corporation.
For the separation calculated by transient dynamic analysis for certain spent fuel transportation casks subject to significant delayed-strike secondary impact effects, momentary gap opening, which may last less than a few milliseconds, has been observed. Gap openings for those casks generally correspond to elastic deformations of fairly long closure lid bolts. Since the mating surface will remain in contact after cask free drops, a leak tight containment boundary is assured, and there is no compelling need for evaluating compression set performance of the seal.

e. Aerosol concentration in the cavity of the package: what are the benchmark values in the U.S.?

NRC Response:
Based on source term it is generally considered to be homogeneous distributed within the cavity for transportation based on a mass fraction of fuel released as 3 E-5. For storage we have considered settling effect which reduces the release rate by an order of magnitude. The fraction of rods that are assumed to fail are 3% and 100%, for normal and accident conditions, respectively.

f. Release rate of fission gases: what are the benchmark values in the U.S.?

NRC Response:
Generally the fraction of rods that develop cladding breaches are assumed to be 30% except for some early BWR fuels which could be as high as 70%

g. Leaktightness of irradiated fuel elements - fuel failure rate: what are the benchmark values in the U.S.?

NRC Response:
The NRC will accept 3% in normal conditions and 100% in accident conditions.

6. Radiation protection

    a. Control of the dose rates before shipment: how are the measurement uncertainties taken into account?

    NRC Response:
    The NRC position (ref: NUREG/CR-5569, pg. 96) is that the result of a valid measurement obtained by a method that provides a reasonable demonstration of compliance or of noncompliance should be accepted and that the uncertainty inherent in the measured value need not be considered in determining compliance or non-compliance with a regulatory limit. In practice, during the conduct of a shipment, measurement uncertainties are not accounted for in the pre-shipment measurements (i.e. the measurements obtained from direct readings are used as read). There can be numerous uncertainties in the measurements such as proper positioning of the detector (which differs depending upon the detector and how the detector was calibrated) and how the proper distance from the package was established. However, all the uncertainties combined won't amount to much of an effect. That is because the shipments are typically at dose rates that are significantly below the dose rate limits. In some cases, NRC inspectors may find a different dose rate than what the shipper measures on a package, but it is also extremely rare to find a maximum dose rate that exceeded the limit when the package user's measurements had shown compliance with the limit.

    For the technical evaluation of CoC applications NRC staff have begun to look more at the uncertainties in the calculations. In each case, the analysis shows that the calculated dose rates were close to the limits (the difference was less than the reported analytical uncertainties). Some people make the claim that calculations may be more trusted than measurements. Regarding pre-shipment measurements, the NRC position is that measurements are to be done with properly calibrated equipment that is appropriate to the radiation type and energy range being measured. In Chapter 7, "Package Operations" of applications (which is incorporated into a CoC by reference) there is a statement that the dose rates are measured and confirmed to comply with regulatory limits prior to being sent out the door.

7. Presence of unexpected objects in the cavity of package

    a. Are some procedures to avoid such unexpected objects?

    NRC Response:
    NRC staff review the operating instructions proposed by the applicant, to ensure that only the contents which are reviewed are placed into the package. We rely on the facility operation and quality control to ensure that there is a program to comply with the requirements of the loading procedure. For our largest spent fuel containers, the utilities are required to have "foreign material exclusion" procedures as a part of the loading, to ensure that objects are not inadvertently dropped into the container. There have been some instances where foreign objects have been accidentally dropped into storage casks, and the utility is required to evaluate the condition. If these storage containers are to be placed into transport in the future, than any foreign materials will need to be evaluated to ensure the container will comply with the transportation Certificate of Compliance.

8. SAR Assessment process: what is the current practice in the U.S.?

    NRC Response:
    The technical review is as follows. SAR is assigned to a project manager, criticality, materials, structural, thermal. shielding, and containment reviewer. Each reviewer reviews the SAR per the Standard Review Plan, NUREG-1609, "Standard Review Plan for

Transportation Packages for Radioactive Material" or NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." If there are questions then the reviewer writes a Request for additional information (RAI) indicating what information is requested, why it is requested, and which regulation requires the information to be met. These are sent to the applicant for response. Based on response the reviewer drafts a safety evaluation report (SER) indicating what was reviewed, what the staff found and what parts of the regulation 10 CFR 71 was met by the applicant.

9. Emergency procedure: what is the current practice in the U.S.?

NRC Response:
The NRC does not review emergency response procedures as a part of the transportation package review. The U.S. Department of Transportation has regulatory requirements for transport carriers and for shippers of hazardous materials packages to develop emergency response information and to maintain this information at all times that the material is in transit. For many shipments, the North American Emergency Response Guidebook pages are used by the shipper and carrier, but the shipper may also develop their own information and provide it to their carrier.

10. Lifting of packages – Brutal putting down: what is the position of the NRC on this subject?

NRC Response:
The 10 CFR Part 71.45(a) requirements, which govern package lifting, do not recognize the "brutal putting down" operating condition. The IAEA TS-R-1, Section 607, provision on snatch lifting, however, is implemented in the U.S. primarily for lifting heavy packages, such as spent fuel transportation casks, which could involve use of building cranes of nuclear power station facilities. A dynamic load factor ranging from 0.1 to 0.2, consistent with applicable crane design standards, is generally considered in determining an "apparent dead load" for which, a minimum safety factor of three against material yield strength must be demonstrated for the lifting device that is a structural part of the package.

11. Drying of the packages: Is there any drying criterion?

NRC Response:
Some considerations for the "drying" of packages:
Transportation packages can have a variety of contents. If the content is radioactive waste, the waste is generally shipped dry (minimize moisture) in order to minimize the effect of radiolysis and chemical reaction. The effects of radiolysis and chemical reaction can be the generation of gaseous products and, in particular, combustible gases (i.e., hydrogen) that must be limited to less than 5% volume. Gases and vapors also should be minimized in order to minimize package pressure at the high temperatures associated with NCT and HAC. In addition, moisture should be minimized in order to prevent "… significant chemical, galvanic, or other reaction among the packaging components, package contents, …" (71.43(d)).
Spent fuel should not be exposed to air or water vapor and so the package should go through a drying type process and then be backfilled with inert gas.
Prior to transport operation (criteria for your references)
Vacuum drying (usually used for burnup < 45 GWD/MTU)
1) hold pressure less than 3 torr for more than 30 minutes.
2) full water in annulus (if annulus exists).

> *1) Forced Helium Drying (usually used for burnup > 45 GWD/MTU) Package gas temperature exiting the demoisturizer shall be < 21°F for > 30 minutes or gas dew point exiting the package shall be < 22.9°F for > 30 minutes. This should ensure the partial pressure of water vapour in the MPC cavity will not exceed 3 torr.*
> *2) No water is allowed in annulus to enhance Forced Helium Dehydration System performance if an annulus exists.*