orano

TN Americas LLC

Columbia Office
7135 Minstrel Way
Columbia, MD 21045
Tel: (410) 910-6900
@Orano_USA

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ATTN: Program Management Announcements and Editing Staff

Subject: Docket ID NRC-2019-0132, TN Americas LLC Responses to Draft NUREG-2216 Standard Review Plan for Spent Fuel Transportation

References: [1] Nuclear Regulatory Commission, Docket ID NRC-2016-0179, Standard Review Plan for Spent Fuel Transportation

[2] Draft NUREG–2216, Revision 0, "Standard Review Plan for Spent Fuel Transportation," Accession No. ML19214A229

TN Americas LLC (TN) herein submits comments for Draft NUREG–2216, Revision 0, "Standard Review Plan for Spent Fuel Transportation" [2].

Should NRC staff have any questions, please contact Mr. Peter Vescovi by telephone at 336.420.8679 or by e-mail at Peter.Vescovi@orano.group.

Sincerely,

Don Shaw

Don Shaw
Licensing Manager
TN Americas LLC

cc:

Prakash Narayanan, TN Americas LLC
Raheel Haroon, TN Americas LLC
Peter Vescovi, TN Americas LLC
Glenn Mathues, TN Americas LLC

Enclosure:

1. TN Americas LLC Comments for Draft NUREG–2216, Revision 0

Enclosure 1 to E-55271

TN Americas LLC Comments for Draft NUREG–2216, Revision 0

Section/Page	Comment
Title, page i	Title should be *NUREG-2216, "Standard Review Plan for Transportation Packages for Radioactive Material."* NUREG-2216 is combining both NUREG-1609 and 1617 into one SRP that is for all types of radioactive material packages. From page xxvi , lines 4-12, of NUREG-2216 – *The SRP is organized to correlate with the recommended content for an application, as detailed in RG 7.9. The individual sections of each chapter address the matters that are reviewed, the basis for the review, how the review is accomplished, and the conclusions that are sought and follow a common outline of subsections, as described below. In conjunction with the SRP, the NRC staff developed several interim staff guidance (ISG) documents related to package approvals under 10 CFR Part 71. An ISG addresses emergent review issues. This SRP combines and updates NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," issued September 1997, and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," issued March 2000, and their supplements and incorporates applicable ISGs, as shown in Table 1.*
ABSTRACT	
Page iii	None
TABLE OF CONTENTS	
Page v	Add GLOSSARY and UNITS.
Page vii	Section numbering in SRP should be consistent with RG 7.9. There is not a separate section in RG 7.9 for Materials Evaluation or Quality Assurance Evaluation. Periodic Review of RG 7.9 done in 2016 (ML18058B996 - Periodic Review of RG 7.9. (2 page(s), 7/18/2016) states that the staff will revise RG 7.9 when NUREG-1609 and NUREG 1617 are combined into a single document.
LIST OF FIGURES	
Page xi	None
LIST OF TABLES	
Page xii-xiv	None
ABBREVIATIONS AND ACRONYMS	
Page xv-xvii	None
UNITS	
Pages xix to xx	None
GLOSSARY	
Page xxii	Add definition of Package Application (Safety Analysis Report) from ISG-20.

Section/Page	Comment
	Package Application (Safety Analysis Report) In the context of Part 71, the Safety Analysis Report (SAR) is called the package application. The sections identified above are typically incorporated by reference into the package approval. Other information provided in the package application report is not typically considered a condition of the approval. The package application simply provides the information that demonstrates that the design meets the performance standards in the regulations. The package application is typically listed as a "reference" at the end of the certificate, not as a condition. To use a package under the General License in Subpart C of 10 CFR Part 71, the licensee is required to have a copy of the packaging drawings and other documents, referenced in the Certificate, that relate to the use and maintenance of the package, and actions to be taken before shipment. The licensee must follow the terms and conditions in the certificate, i.e., the shipment must conform, in all respects, to the certificate and any documents specifically cited as a condition of the approval. The licensee does not need to have a copy of the complete package application. This is in contrast to casks licensed under 10 CFR Part 72, where a general licensee must have, and is required to review, the SAR, including updates (see, for example, 10 CFR 72.212(b)(3) and 72.248(c)(7)).
INTRODUCTION	
Page xiii of the table of contents	, Tables: Table x-1 for each chapter, “Relationship of Regulations and Areas of Review for Transportation Packages.” Consider a consistent layout with the 10 CFR 71 regulation addresses along the "y" axis and the areas of review along the 'x" axis. There appears to be far fewer areas of review versus the 10 CFR 71 regulations associated with the areas of review. See page 1-3, Table 1-1 as a good example. From reviewing all the x-1 tables, there was inconsistency in the table layout.
Page xxvi (Lines 4 and 5)	The sentence states “The SRP is organized to correlate with the recommended content for an application, as detailed in RG 7.9.” However, Materials is discussed within Chapter 2 and Quality Assurance is not in RG 7.9. Need to address these differences.
1 GENERAL INFORMATION EVALUATION	
Page 1-3, Table 1-1 (and similar Table in other sections):	NUR\EG-1617, Appendix A-Stanard Review Plan Correlation with 10 CFR PART 71 and Regulatory Guide 7.9 should be retained in NUREG-2216 as an Appendix. The Table 1-1 provided in 1.3 Regulatory Requirements and Acceptance Criteria should be with the table provided in NUREG-1617 Appendix A. The correlation to RG 7.9 missing from all sections in NUREG-2216. This comment applies to all Sections 1 through 10 of NUREG-2216.
Page 1-15 and 1-16 (lines 39, 41, 3, 6, 8, 11, 12, and 14):	The numbering-scheme for “Evaluation Findings” is not consistent with the other sections. Bullet symbols used in Section 1.5 Evaluation Findings instead for; F1-1, F1-2,.., F-n.
Page 1-16 (lines 21-23):	There is not a list of RG and NUREGS that provide guidance lists in Section 1,3,4,5,6,7,8,9, and 10, as is done in Section 2-Structural Evaluation, under heading 2.4.1.2 Identification of Codes and Standards for Package Design. There should be a parallel structure to

Section/Page	**Comment**
	providing applicable RG and NUREG generic guidance in each section of SRP as is done in 2.4.1.2.
2 STRUCTUAL EVALUATION	
Page 2-9 (lines 32-34)	*2.4.4.1 Evaluation by Analysis* *Should refer to structural analysis instead of thermal analysis*
Page 2-13 (line 32)	Missing parenthesis after abbreviation for pressure..Standard convention is to use absolute after the units and use abbreviation psia. (Reference ASME ANSI Y14.38- Abbreviations for Use on Drawings and in Text) “external pressure equal to 25 kilopascals (kPa) (3.5 pounds per square inch absolute (psia)) ~~absolute~~ as”
Page 2-19 (line 1)	Replace SI pressure units with abbreviation for megapascal to be consistent with use of abbreviation psi for pounds per square inch. “2 MPa ~~megapascals~~ (290 psi) for a period of not less than 1 hour without collapse, buckling, or”
Page 2-21 (line 33)	“Society of Mechanical Engineers (ASME) Boiler and Pressure (B&PV) Code, 2007—Addenda 2008.” Why is 2007-addenda 2008 code year specified?
3 THERMAL EVALUATION	
Page 3-3, Table 3-1	List 71.51(a) regulation relationship for "Description of the thermal design" to be consistent with Section 3.3.1 (see Page 3-4, Line 10)
Page 3-3, (line 3)	Add "Normal Conditions of Transport," before "primarily", similar to “Hypothetical Accident Conditions” listed in Line 4 “b. 10 CFR 71.71, "Normal Conditions of Transport," primarily 71.71(c)(1) and 71.71(c)(2), for SNF packages.
Page 3-5, (line 1)	Add Reference source [10 CFR 71.71(b)] at the end of "...under consideration." for normal conditions of transport. “which is the most unfavorable condition for the feature under consideration [10 CFR 71.71(b)].”
Page 3-5, (line 23)	Add Reference source [10 CFR 71.73(b)] at the end of "...under consideration." for hypothetical accident conditions. “feature under consideration [10 CFR 71.73(b)]. The initial internal pressure within the containment system must be”
Page 3-13 (lines 8-11)	*3.4.3.1 Evaluation by Analysis* *Should refer to thermal analysis instead of structural analysis*
Page 3-18, (lines 3-14)	Should use "extend below" to replace "exceed" for minimum temperature limits. . It is not accurate to say "not exceed minimum allowable limit for the minimum temperature" “Confirm that the maximum ~~and minimum~~ temperatures do not exceed

Section/Page	Comment
	their allowable limits and minimum temperature do not extend below their allowable limites, as 14 specified in Section 3.4.2.3 of this SRP chapter."
Page 3-27, (line 13)	Using "short term" to replace "short time" as consistent with "short-term operations" referred in Page 3-2, Line 7. "the specified allowable short term ~~time~~ limits during hypothetical accident conditions"
4 CONTAINMENT EVALUATION	
Page 4-6, (line 9):	"Value" is incorrect and needs to be "Vessel." "applicable codes and standards (e.g., American Society of Mechanical Engineers Boiler and Pressure Vessel ~~Valve~~ Code specifications for the vessel)"
Page 4-11, Table 4-2.	There should be NUREG/CR-6487 source document referenced for values in this table.
Page 4-10, (lines 10 to 33)	It is not clear if this approach applies to non-fuel hardware (NFH). For instance, does crud form on a BPRA similar to fuel? What fraction of helium would be released from a BPRA cladding breach (pressure rise)? This section should acknowledge NFH and provide guidance. See also page 4-12, lines 20 and 21, where breaches relate only to "fuel rods."
5 SHIELDING EVALUATOIN	
Chapter 5, general	Chapter 5 is written from a "perfect world" perspective it which is it assumed that all possible information is known about all radioactive items to be shipped. However, the "real world" will not always be so clear. Given the age and uncertainty involved with material currently loaded into canisters, some of which were loaded over 20 years ago, this approach has two highly probable outcomes: (1) material is inadvertently shipped in violation of the CoC, or (2) an NRC amendment is sought for every new client, as each potential client may have material that does not exactly fit the CoC. The NUREG should provide a roadmap of what to do if material to be shipped does not exactly match the CoC requirements but is reasonably close, or if the material has properties that are not well documented.
Page 5-9, (line 5)	Clarify that if the impact limiters are wider than the bed of an open transport vehicle that the vehicle limit (200 mrem/hr) applies at the vertical projected surface of the impact limiter radius, and 2 m may be measured from the impact limiters .

Section/Page	Comment
Page 5-13, (lines 5 and 6)	Requiring “maximum activity” to be provided is a complex question with little value if the package is leaktight. ORIGEN can provide hundreds to thousands of radionuclide concentrations, most of which have no relevance to shielding. It is not clear which radionuclides the NRC wishes to see. Only ~10 radionuclides are typically relevant to shielding if the system is leaktight. To further complicate matters, radionuclide concentrations are a complex function of burnup, enrichment, and cooling time (BECT), so developing maximum activities across all possible combinations is a significant undertaking. Bounding BECT combinations are typically determined without studying the nuclide activities.
Page 5-13, (lines 42 and 43)	Different NRC reviewers have vastly different interpretations of what is reasonable in regards to cladding degradation in high burnup fuel. The NRC, in conjunction with PNNL and/or ORNL, should provide a clear recommendation here. The NRC and the national labs, not private industry, are the experts in this area.
Page 5-14,(lines 6-19)	Provide a reference for all claims regarding axial blankets and effects on source terms/dose rates. TN Americans has had extensive discussion with the NRC on this topic, and our NRC reviewer takes a different position than what is written here. The key claim that needs a reference is: “However, the impact is insignificant for natural uranium blankets shorter than 15 cm (6 inches)”. Also, “insignificant” needs to be quantified, especially when the limiting dose rate is typically 10 mrem/hr @ 2 m. Keep in mind also that if a source term is developed for the average assembly enrichment, the ORIGEN run will be performed for an enrichment lower than the “middle” part of the fuel assembly. Because lower enrichments increase the neutron source, modeling a lower enrichment counteracts part of the neutron source increase due to the higher burnup. The NUREG should discuss this effect. In addition, because the gamma source is proportional to burnup, a fuel assembly with axial blankets should not have a larger gamma source, contrary to what is written in the NUREG.
Page 5-14, (lines 41 and 42)	Would “Co-60 equivalent” be an acceptable means to characterize non-fuel hardware (NFH) (e.g., BPRA) in the CoC? In this manner, the NFH analysis could be performed using Co-60 and applied to any NFH, including silver-indium-cadmium or hafnium NFH. Of course, this would require a characterization document for all NFH to be shipped and a method to determine Co-60 equivalence. See also page 5-18, lines 10-22, for the difficulty in characterizing NFH sources. Also, page 5-19, lines 5 and 6, for a discussion of cobalt activity and other NFH types .
Page 5-16, (lines 8 and 9)	NUREG should state that if SCALE6 is used to develop source terms for standard BWR and PWR fuel that this reference documentation is not needed.

Section/Page	Comment
Page 5-16, (lines 20 and 21)	NUREG should state what is an acceptable uncertainty for SCALE6 source terms. Our experience is that some NRC reviewers reduce the 2 m dose rate to < 9 mrem/hr, which does not seem reasonable when using an NRC-developed code. NRC should coordinate with ORNL and provide a clear recommendation.
Page 5-18, (lines 35 to 44)	In the mid-1990s, due to a cobalt reduction program, the maximum cobalt impurity was reduced to approximately 500 ppm. The NRC, in conjunction with ORNL, should provide a clear recommendation here on what to assume pre-1995 and what to assume post-1995. PNL-6906 is a very old reference. Old references are also available from ORNL that state the impurity in steel is 800 ppm. This level of uncertainty can cause much difficulty when licensing a package, especially if each NRC reviewer has a different opinion on what is reasonable .
Page 5-22, (lines 38 and 39)	"Credible and bounding" has different meaning for different NRC reviewers. More concrete guidance for "standard" high burnup fuel should be provided.
6 CRITICALITY EVALUATION	
Page 6-18, (lines 11 and 12)	What if some water between the packages in the NCT array is more reactive than void? The wording of this bullet seems to imply that cases with water between the packages is not needed for the NCT array, even if more reactive.
Page 6-26, (lines 12 thru 14)	"Bias values <u>should be</u> added to the calculated package k-eff, while bias uncertainty values <u>may be</u> statistically combined with other independent uncertainties." The language is rather imprecise. If k_{eff} + 2 ≤ USL, does this mean we can *either* add the bias uncertainty to k_{eff} *or* include with ? Si penalizing to combine the bias uncertainty with "other independent uncertainites," especially since the Monte Carlo uncertainty is usually very small. However, it does not appear that combining the bias uncertainty with is require recommending.
Attachment 6A	In multiple places, ISG-8 Rev. 2 is referenced, although Rev. 3 is the latest.
Page 6A-28, lines 8 through 11	The RW-859 database was updated in 2013.
7 MATERIALS EVALUATION	
Pages 7-1 thru 7-42	None
8 OPERATING PROCEDURES EVALUATION	
Page 8-6, (lines 9 thru 11)	Operating instructions should consider other causes for exceeding expected external radiation levels measured prior to transport. In addition to a improperly loaded package, unexpected external radiation measurement could be caused by a trespass radiation from a nearby source term or damage to the cask..
9 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM EVALUATION	
	None

Section/Page	**Comment**
10 QUALITY ASSURANCE EVALUATION	
Pages 10-1 thru 10-18	Quality Assurance Program Descriptions (QAPD) in the Package Application (Safety Analysis Report) are not package specific, and NRC issues Quality Assurance Program Approval Certificate to the certificate holder. There is no regulatory requirement to include the QAPD. Section 10 of the SRP should not be part of the SRP.
APPENDIX A DESCRIPTION, SAFETY FEATURES, AND AREAS OF REVIEW FOR DIFFERENT TYPES OF RADIOACTIVE MATERIAL TRANSPORTATION PACKAGES	
Pages A-2, A-8, A-13 thru 15, , A-20, A-24, A-28, A-33, A-37, A-41.A-47,	Listing of Appendices A through E. Improve the quality or update the various types of transportation packagings. These appear such that there were not any updating to the descriptions, etc., especially the quality and orientation of the figures within each appendix.
APPENDIX B DIFFERENCES BETWEEN THERMAL AND RADIATION PROPERTIES OF MIXED OXIDE AND LOW-ENRICHED URANIUM RADIOACTIVE 3 MATERIALS	
Page B-1 thru B-7	None
APPENDIX C DIFFERENCES BETWEEN THERMAL AND RADIATION PROPERTIES OF MIXED OXIDE AND LOW-ENRICHED URANIUM SPENT NUCLEAR FUEL	
Page C-1 thru C-11	None
APPENDIX D BENCHMARK CONSIDERATIONS FOR MIXED OXIDE RADIOACTIVE MATERIALS AND SPENT NUCLEAR FUEL	
Page D-1 thru D-11	None
APPENDIX E DESCRIPTION AND REVIEW PROCEDURES FOR IRRADIATED TRITIUM-PRODUCING BURNABLE ABSORBER RODS PACKAGES	
Page E-1 thru E-39	None