ENERGYSOLUTIONS

15 May 2012
EL&P-016-12

Mr. Pierre Saverot
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT:	Amendment Request for Certificate of Compliance No. 9168 for the Model 8-120B Package - Request for Supplemental Information
Docket No.	71-9168
TAC No.	L24549

Dear Mr. Saverot:

Energy*Solutions* provides the attached response to the second request for additional information (RAI) dated 28 March, 2012.
In addition to specific responses to the RAI items, EnergySolutions provides revised SAR Chapters 1, 4, 5, 7, and 8 incorporating the information provided in our response. Please replace the previously submitted Chapters, with the Chapters attached to this letter.
Should you or members of your staff have any additional questions about the response, please contact me at (803) 758-1898.

Sincerely,

Mark Whittaker
Sr. Health Physicist, Radiological Services

Attachments:

- Response to RAI#2
- SAR Rev.1, May 2012 Chapters 1, 4, 5, 7, 8
- References, Documents, and Data (Proprietary and Non-proprietary, provided on CD/DVD)
- Proprietary Affidavit

NMSS01

Chapter 1 – General Information
Licensing Drawings and Bill of Materials

1-1 Clarify and limit what corrective actions will be taken if the lead shielding has a 10 percent or greater loss of material.
The corrective remedies mentioned in note 6 on sheet 1 of Licensing Drawing C-110-E-007, Rev. 16, are not specified. In previously approved applications, limited quantities of lead metal (the amount was specified on the licensing drawings) were permitted to be attached to the gamma shield to mitigate casting defects. The re-melting of lead while still in the annulus (presumably through a high-temperature heat source such as a blowtorch) could affect the properties of the steel shell and may not be acceptable.
This information is required to demonstrate compliance with 10 CFR Part 71.33(a)(5) and 71.47(a).

Response:
SAR Drawing C-110-E-007, Note 6 will be revised as follows:

NOTE 6: Lead and shielding effectiveness: All lead pouring shall be done continuously to avoid gaps or radiation streaming paths. Gamma Scan is required to ensure shielding integrity. Measurements indicating lead layer of less than the minimum thickness specified shall require remedy. Gamma scan methodology and remedy limitations are specified in Chapter 8 of SAR. The procedures and equipment to be used for the Gamma Scan shall be approved by EnergySolutions prior to performance of the test. Gamma Scan is QL-2.

In addition, the following paragraph will be added to Section 8.1.5:

Remedy for an unacceptable gamma scan include actions such as controlled re-heating of the cask body to melt the lead to remove any voids or streaming paths. This process may be used as long as average metal temperatures are kept below ~800°F.

Chapter 2 – Structural Evaluation

2-1 Justify the lead slump amount of 0.15 inches, and the punctured lead thinning of less than 0.5 inches resulting from the Hypothetical Accident Conditions (HAC) tests.
The staff could not verify these numbers from calculations presented in ST-627 Rev. 1, nor from calculations in ST-679 Rev. 0. These values were further used in the modeling of the shielding of the package. Provide details of how these numbers were derived, and indicate where these numbers can be found in each of the two calculations mentioned above.
The information is required to demonstrate compliance with 10 CFR 71.71 and 71.73.

Response:
In Section 2.7.1.5 of the SAR it is stated, "From the finite element model analysis the relative displacement at the lead-steel interface is obtained. Figure 2-50 shows the exaggerated displacement plot under this drop condition. The total relative displacement of the lead column (0.141 inch) is reported as the lead-slump".
Figure 2-50 shows the total deformation of the cask during the end drop. The relative displacement of the top of the lead-column and the bottom of the bolting ring is the lead-slump. The data print-out from ANSYS for this load case is included in Reference 2-15 (ST-627) Appendix 2, Page 24 of 24. The node representing the top of the lead column is 732 and that of the bottom of the bolting-ring is 14,402. The difference of the total displacements of these nodes

is the relative displacement. The calculated lead slump (relative displacement) of 0.14057" is also indicated on this page.
In Section 2.7.3 of the SAR it is stated, "A conservative evaluation of the maximum amount of lead deformation under puncture drop test of 8-120B cask has been performed in Reference 2-15. It has been shown that the lead shielding deformation is limited to 0.458 inch".
This evaluation can be found in Section 7.5 Pages 17 through 19 of Reference 2-15 (ST-627).

Chapter 5 – Shielding Evaluation

5-1 Modify the first paragraph of Section 1.2.2.2 of the application to remove the contradiction regarding neutron producing materials in the contents.
In response to the previous RAI 5-1, the applicant added a statement to the first paragraph in Section 1.2.2.2 to preclude neutron-producing materials. The statement, however, introduces a contradiction to the sentence preceding it allowing fissile materials up to the fissile exemption limits to be present in the package. Fissile materials produce neutrons. Thus, the paragraph should be modified to remove the contradiction. One possible option is to rephrase the last sentence of the paragraph to read as: "Materials producing neutrons through α,n or γ,n reactions and materials, other than fissile materials as allowed in the preceding sentence, producing neutrons through spontaneous fission are not authorized." The modification should ensure that the neutron source of the contents remains insignificant; otherwise, the shielding evaluation should be modified to account for neutron sources.
This information is required to demonstrate compliance with 10 CFR 71.33(b), 71.35(a), 71.47, and 71.51.

Response:
Section 1.2.2.2 has been revised as suggested.

5-2 Provide the information requested in the previous RAI 5-4.
In the previous RAI 5-4, staff requested additional information regarding shipment data to be used to support the applicant's shielding evaluation. As has been indicated in a previous review (see the SER for the recent Model No.10-160B package review) and as evidenced by the RAIs associated with this current application, staff continues to have concerns regarding the shielding evaluation and contents specification method proposed by the applicant for this package. Dose rate measurement data from previous shipments with the Model No. 8-120B package can provide significant support in demonstrating the adequacy and/or conservative nature of the proposed evaluation and contents specification method. The use of the data may alleviate the need for further significant scrutiny of the method. However, the staff finds that to be sufficient for this purpose the data needs to include the information requested in the previous RAI 5-4. The RAI does not request a statistically significant sampling; rather, it requests a sampling that can be reasonably construed to represent the contents to be shipped in the Model No. 8-120B package with respect to the contents characteristics described in that RAI. The adequacy of the radiation surveys in identifying the locations of and the values of the maximum dose rates for the package should also be justified.
This information is required to demonstrate compliance with 10 CFR 71.35(a), 71.47, and 71.51.

Response:
Energy*Solutions* provides the attached set of data (CD) from 8-120B shipments made to our Barnwell disposal facility. The attached data set includes all shipments made to Barnwell

between 2008 and 2011 and includes all the radiological data held by Energy*Solutions* concerning the shipment. The identity of the shipper is redacted. Data included for each shipment includes the NRC 740 and 741 forms and the receipt survey performed at Barnwell. Also included, if provided by the shipper, are surveys made prior to shipment. Energy*Solutions* has no knowledge of the procedures for performance of the shipper surveys nor control of what data in addition to the 740 and 741 forms is provided by the shipper. A summary spreadsheet is included which lists the shipment number, receipt date, TI, and waste description.

5-3 Modify the method for determining the allowable gamma source and the procedures in Attachment 1 to Chapter 7 of the application so that the 5 percent reduction is performed as part of the procedures performed by the package user to determine the allowable contents.
Staff finds that reducing the allowable source strengths, as discussed by the applicant in its response to the previous RAI 5-6, is an appropriate way to account for uncertainties and non-conservatisms in various areas of the range of applicability for the method to determine the allowable package contents. However, staff finds that the application of the 5 percent reduction is more effective when incorporated as part of the procedures performed by the package user considering the scale of values in Figure A-1 of Attachment 1 and the level of precision that the figure allows. A 5 percent reduction in the curves themselves is nearly indistinguishable and will not lead to any real difference in the allowable source strength of the contents that would be determined by the entity that uses the figure.
This information is required to demonstrate compliance with 10 CFR 71.47, 71.51, and 71.87(a) and (f).

Response:
A 5% conservatism factor has been included in the payload qualification process described in Chapter 7 by requiring the "sum-of-fractions" be less than 0.95.

5-4 Provide further justification for the need to address bremsstrahlung only for total beta source activities exceeding $2x10^{12}$ betas/second.
In response to the previous RAI 5-8, the applicant provided a method to address bremsstrahlung from beta-emitting contents. As part of that method, the applicant proposed to only require consideration when the source strength exceeds $2x10^{12}$ betas/second. While the applicant provided a basis for deriving this value, it is not clear that a lesser source strength need not be considered. For example, for a point source with betas with a maximum energy of 3.5 MeV, this source strength (assuming the average beta energy is one third the maximum) would result in a fraction of about 0.106 for a ratio of the equivalent gammas to the allowable gamma strength. Since the HAC case is limiting for point sources, this translates to 106 millirem per hour (mrem/hr) at 1 meter (or about 10 percent of the dose rate limit). Staff finds that this could potentially result in a package that exceeds the allowable limits and that, therefore, a lower threshold is warranted. Staff recognizes that the applicant compared dose rates between the equivalent gamma source and the beta source, both calculated using MCNP. The results indicate significant conservatism in the method. However, these calculations appear to be for normal conditions of transport (NCT) only. The currently proposed threshold beta source strength may be found acceptable if the applicant can show a similar level of conservatism for a point source under HAC, including for a maximum beta energy at the maximum gamma energy allowed in the package.
This information is required to demonstrate compliance with 10 CFR 71.47, 71.51, and 71.87.

Response:

To address the concern, we modified the package HAC MCNP model by replacing the photon source with a $^{90}S/^{90}Y$ beta source spectrum, and rerunning the model in e-p transport mode to simulate the production of bremsstrahlung radiation. ^{90}Y betas are very energetic daughters of ^{90}Sr, with a maximum energy of 2.245 MeV. The bremsstrahlung response for a unit beta source was 2.3E-12 mrem/hr per β/sec. So for a $^{90}S/^{90}Y$ $2x10^{12}$ β/sec source, the HAC dose rate at 1 m due to bremsstrahlung would be 4.6 mrem/hr, or 0.5% of the HAC 1 m dose limit. Examination of radionuclides with beta energies significantly higher than that of Y-90 shows that these radionuclides have very short half-lives and are thus not likely to be found in the contents of the 8-120B. This demonstrates the reasonableness of the proposed qualification methodology requiring only beta sources greater than $2x10^{12}$ β/sec to be qualified.

5-5 Clarify the following statements:
a. Shoring is used to prevent axial movement of contents as well as radial movement.
b. Shoring is only credited in NCT for maintaining position and not for providing shielding.
This information is important to understanding the applicant's evaluation method for the allowable contents and its relation to how the package is operated.
This information is required to demonstrate compliance with 10 CFR 71.47, 71.51 and 71.87.

Response:

The revised shielding evaluation, described in the revised Chapter 5, includes certain source/cask geometries that rely on the radioactive material being shored at the cask centroid during NCT. This shoring prevents both axial and radial movement during NCT but does not provide attenuation, i.e., shielding, in the shielding analysis

5-6 Justify the following statements, modifying the evaluation as necessary:
a. The evaluation method adequately accounts for the variation in distribution of radioactivity within the source volume for distributed sources for NCT and HAC.

Response:

The shielding analysis methodology has been substantially changed, in a way that fully addresses any potential variation of radioactivity concentration within the source (or payload). All references to a "uniformly distributed" (or, "essentially uniformly distributed") source have been removed, and any criteria associated with source uniformity (e.g., allowable factor of three variations, etc.) are no longer applicable.

Limits on the gamma source strength density within the payload material (in gammas/sec per gram of source material) are determined by analyses which conservatively model the entire cask cavity as being filled with material at the limiting source strength density. To qualify for loading, the maximum gamma source strength density that occurs anywhere within the payload material(s) must fall under the calculated limit. Thus, the analyses effectively treat a payload with spatially varying radioactivity concentrations (or gamma source strength densities) by conservatively modeling the "hottest" section of the payload throughout the entire cask cavity. This produces bounding cask exterior dose rates, at all cask exterior locations.

The shielding analyses also demonstrate that cask exterior dose rates primarily scale with the gamma source strength density (gammas/sec per gram) within the source material (as opposed to the overall source strength in gammas/sec, or the volumetric source strength density, in gammas/sec-cc), and that for a given gamma source strength density (in gammas/sec-gram), modeling an upper-bound source region material density is conservative.

Special cases where smaller volumes of waste are shored to the centroid of the cask cavity (under NCT) have also been evaluated. In these cases, the limiting gamma source density (in gammas/sec-gram) is applied uniformly over the entire specified source region volume (which covers a fraction of the cask cavity, near the cavity center) in the NCT shielding analyses. Due to the smaller waste (source) volume, and its location at the cask cavity center, these configurations allow higher gamma source strength densities (gammas/sec-gram).

Additional analyses, which determine limits on the overall gamma source strength (gammas/sec) within the cask cavity, conservatively model the entire source strength as a point source. Concentrating the source to one location yields higher peak dose rates on the regulated cask exterior surfaces. No materials are modeled within the cask cavity in these analyses, so all waste self-shielding effects are conservatively neglected. These analyses also place the point source at a worst-case location within the cask cavity. An exception to this are the analyses used to determine source strength (gamma/sec) limits on small payloads that are shored to the centroid of the cask cavity (for which a cavity-center point source is modeled). Given that these analyses concentrate the entire source strength within the cask cavity to a single point, issues associated with the distribution of the source strength within the source material are not applicable, and the analyses clearly bound any actual configuration.

b. The distribution of radioactivity within the source volume won't change due to HAC.
In response to the previous RAI 5-5, the applicant indicated it had changed one of the criteria for a distributed source to be the source that must meet the definition of "essentially uniformly distributed." It is not clear from the application how the evaluation accounts for the distribution of radioactivity within the source volume that is allowed by this definition under NCT and HAC. The evaluation appears to rely on a uniformly distributed source, whereas the distribution can vary by a factor of three. Also, the applicant indicated that the distribution of radioactivity within the source volume would not be affected by HAC. The basis for this statement is not clear. It is not clear that the definition of "essentially uniformly distributed" applies to a package that has experienced HAC. Thus, the applicant should justify the statement. The evaluation should be modified, as needed, to address these concerns. Staff calculations indicate there may be concerns for meeting dose rate limits when accounting for variations in the source distribution.
This information is required to demonstrate compliance with 10 CFR 71.47 and 71.51.

Response:

Given that source strength density averaging is no longer used, any redistribution, or concentration of the "hottest" components of the payload under HAC, is no longer applicable. The analyses now conservatively model the "hottest" part (or component) of the waste material throughout the entire cask cavity.

For the cases that evaluate reduced-volume payloads that are shored to the centroid of the cask cavity (under NCT), the HAC analyses still model the entire cask cavity filled with the "hottest" part of the waste (source) material. In other words, the gamma source strength density (in gammas/sec-gram) is limited to a value that would not result in cask exterior dose rates over the HAC limit, even if the entire cask cavity were filled with material at the limiting source strength density. This accounts for any change in position, or shape, of the reduced-volume payload (or source) under HAC, since a cask cavity that is completely filled with limiting source density (i.e., "hottest") material bounds any cask cavity that is partially filled with that same material, regardless of the (partial) payload's position within the cavity.

Similarly, the shielding analyses that determine the limits on gamma source strength (in gammas/sec) for the reduced-volume, centered payloads, which model point sources at the center of the cask cavity for NCT, move the point source to a worst-case location within the cask cavity for HAC. This accounts for any repositioning of the source, to any location within the cask cavity, and that may occur under HAC.

5-7 Justify the applicability of the current evaluation to distributed sources that don't fill the whole package cavity, and modify the evaluation as necessary.
In response to the previous RAI 5-10, the applicant discussed smaller distributed sources with increased densities versus the sources filling the entire package cavity. The staff's concern is more with smaller volume sources that have the same (or lower) density and the same source strength as the sources that fill the package cavity. It appears from the procedures described in Attachment 1 to Chapter 7 that these kinds of sources are acceptable. Thus, these kinds of sources should be evaluated for NCT and HAC dose rate compliance. Staff calculations indicate that there may be concerns for meeting dose rate limits for these sources, at least for NCT. Concerns regarding radioactivity distribution within the source described in RAI 5-6 above should also be addressed for these sources too.
This information is required to demonstrate compliance with 10 CFR 71.47 and 71.51.

Response:

For a given, specified, source strength (in gammas/sec) within a distributed source, reducing the volume or material density of the source region may result in higher cask exterior dose rates, due to decreased self-shielding within the source material.

The shielding analysis methodology has been substantially changed. Under the new analysis methodology, the distributed-source analyses determine limits on the gamma source density (in gammas/sec-gram) within the source material. If the gamma source density (in gammas/sec per gram of source material) is held constant, reducing the volume or density of the source material causes cask exterior dose rates to decrease (albeit very slowly, in most cases).

The new shielding analyses that determine limits on the gamma source density within the waste material conservatively assume that the entire cask cavity (or specified, reduced-volume source region) is completely filled with source material at the limiting source strength density. This bounds any smaller source volumes or partially filled cask cavities. The analyses also demonstrate that modeling an upper-bound material density within the source region bounds lower material densities, if the gamma source strength density (gammas/sec-gram) is held constant.

5-8 Provide the following:
a. Verification that the calculations that form the basis of the shielding and allowable contents determination method are correct and yield dose rates that meet the regulatory limits. The proposed contents limits should be adjusted as necessary.

Response:

The shielding analysis methodology has been substantially revised to address NRC concerns and to ensure that all payloads allowed by the loading specification are bounded/covered by the shielding analyses and produce cask exterior dose rates that do not exceed any regulatory limits.

The point source analyses place the (point) source at the worst-possible cask cavity locations (for HAC and/or unshored payload cases), do not take credit for any waste self-shielding effects, and employ small tally regions (1 inch high in key locations) that should detect any local peaks in the dose rate.

b. Justification of the configuration of the model related to the lead slump in the HAC models.

Response:

The response to RAI #2-1 states that the structural analysis shows lead slump due to the drop is 0.141". The value 0.15" has been conservatively included in the HAC shielding model to account for the lead slump.

c. The maximum dimension of the gap between the package top flange and the primary lid, modifying the licensing drawing to include this dimension.

Response:

The drawing has been revised to specify a minimum and maximum thickness of the seal plate (the thickness of the seal plate defines the amount of gap) and remove the gap dimension.

d. Verification that the HAC models account for lead thinning due to the puncture test.
Staff has performed some confirmatory calculations (using the applicant's source configurations) to verify that the allowable contents determined by the applicant's method will meet the regulatory dose limits. These calculations indicate higher dose rates with dose rate limits being exceeded for some of the contents limits, e.g., point surces. Also, the shielding HAC model should account for the full amount of lead slump predicted in the structural evaluation (see RAI 2-1 of this letter).
Additionally, the shielding evaluation and models appear to rely on the gap between the

primary lid and the top flange being a maximum value; this aspect affects the configuration of the lid steel versus the lead and lead slump. The licensing drawing shows the dimension of the gap as a minimum. Thus, it appears the drawing should be modified to show the gap dimension as a maximum value.
Staff looked at the sample inputs provided by the applicant for HAC dose rate calculations. Staff was unable to identify that any of the HAC sample inputs include thinner lead resulting from the puncture test, whereas at least the models used to support Appendix 5 of Reference 5.7.2 were supposed to include this effect. HAC modules shall include lead thinning.
This information is required to demonstrate compliance with 10 CFR 71.47 and 71.51.

Response:

The structural analyses show that local radial lead shielding thinning due to the HAC puncture drop will not exceed 0.5 inches. The HAC shielding analyses conservatively address this potential effect by reducing the thickness of the entire radial lead shield by 0.5 inches.

Chapter 7 – Package Operations
7-1 Modify step 2 of Attachment 1 to Chapter 7 of the application to clarify how a source is determined to be a point source or a distributed source.
In response to the previous RAI 7-5, some clarification was provided regarding this procedure. However, the description is still not adequately clear. One possible option is to modify the third sentence of step 2 of Attachment 1 to read as follows: "Content that does not meet both of these criteria for a distributed source shall be considered a point source." The procedure needs to clearly indicate that contents must meet both criteria (the definition of "essentially uniformly distributed" and the minimum volume) to be classified as a distributed source for the purpose of determining the allowable source strength.
This information is required to demonstrate compliance with 10 CFR 71.47 and 71.87.

Response:
Attachment 1 to Chapter 7 has been completely re-written to incorporate the modified method for determining activity limits described in Chapter 5. The revised Chapter 7 is included in the submittal containing these responses.

7-2 Modify the following:
a. Section 7.0 of the application to indicate that the maximum permissible activity for beta emitting contents is the lesser of 3,000 A_2 and the maximum activity determined per Attachment 1 for packages containing a cumulative beta source strength that exceeds the minimum value justified in the applicant's shielding evaluation.

Response:
Section 7.0 (included) has been revised as requested.

b. The heading of Attachment 1 to indicate the procedure is also used for determining beta activity limits.

Response:
Section 7.0 (included) has been revised as requested.

c. Step 3 of Attachment 1 to Chapter 7 to clarify that the minimum beta source strength for which the procedure applies is a package's total beta source strength.

Response:
Attachment 1 has been completely rewritten and includes the requested revision.

d. Step 3 of Attachment 1 to Chapter 7 to describe the factor that replaces the 9.1E-03 factor for converting the beta source to an equivalent gamma source for contents with significant quantities of materials with atomic number (i.e., Z) greater than 26.
The applicant developed a method for determining the maximum beta source strength for the package in response to the previous RAI 5-8. The applicant also modified the procedures in Attachment 1 to Chapter 7 to include procedures for the package user to determine the allowable beta emitting contents for their particular shipment. However, the description in Section 7.0 (and the heading to Attachment 1) needs to indicate to the user that this procedure needs to be performed for beta-emitting contents. It currently does not; thus, the user may not recognize this condition of package use and improperly load the package.
The procedures in Attachment 1 and the description in Section 7.0 should clearly indicate that the minimum beta strength for which the method applies is the total beta strength for the package (i.e., it is not per-nuclide beta strength); thus, the procedures must be used for packages for which the total beta strength for the nuclides present in the package exceeds the minimum strength described in the procedures. Reference 5.7.2 indicates that the factor of 9.1E-03 is acceptable for contents materials with Z up to 26 and that for contents of a higher Z, the factor should be replaced by 3.5 x Z x E-04. The procedures should include instructions regarding the appropriate factor for use with the contents. This information is required to demonstrate compliance with 10 CFR 71.87(a) and (f).

Response:
Attachment 1 has been revised so that a weighted average Z is determined and used to calculate f using the weighted average Z for all contents.

7-3 Modify paragraph 7.1.19.3 of the application to read as follows: "That the provisions of 10 CFR 71.87 are met, including that the external radiation dose rates are less than or equal to 200 mrem/hr at the surface and less than or equal to 10 mrem/hr at 2 meters in accordance with 10 CFR 71.47 by performing radiation surveys. These surveys should be sufficient to ensure that any non-uniformity in the distribution of radioactivity does not cause the surface or the 2 meter limit to be exceeded." (Italics added to note suggested changes to the text.)
As indicated in the previous RAI 7-2, the currently proposed language is not clear. This information is required to demonstrate compliance with 10 CFR 71.47 and 71.87(j).

Response:
Step 7.1.19.3 has been revised as requested.

7-4 Make the following corrections:
a. The regulatory reference in paragraph 7.3.2 should be changed to 49 CFR 173.428(d); there is no 10 CFR 173.428(e).

Response:

Step 7.3.2 has been revised as requested.

b. The new step 5 in Attachment 1 to Chapter 7 should refer to the new step 4 (and not the new step 3).

Response:
Attachment 1 has been completely rewritten and includes the requested revision.

c. The precision and accuracy of the gamma strength limits in the examples in Attachment A-1 should be consistent with the precision and accuracy allowed by Figures A-1 and A-2 from which the limit values are to be derived by the package user.

Response:
Attachment 1 has been completely rewritten and includes the requested revision.

d. Several of the values for source limits and density correction factors (DCF), in appropriate examples, appear to be inconsistent with Figures A-1 and A-2 and inconsistent with the procedures outlined in Attachment 1 for energies outside the range shown in the figures.

Response:
Attachment 1 has been completely rewritten and includes the requested revision.

e. Some contents activities in the examples also appear to be incorrect or not used in the full process of determining acceptability of the contents.
These questions still remain from the previous RAI 7-3 and as a result of changes made to respond to other RAI questions. Any changes to respond to RAI questions should be reviewed altogether to ensure the resulting operations descriptions are accurate and correct.
Regarding item c, Figure A-2 allows for a precision of 0.05 for the DCF and 0.5 for payload density, while the precision of the limits from Figure A-1 is limited due to the logarithmic scale. For item d, staff noticed that some limit values (and DCF values in applicable examples) don't match the values derived from the attachments figures. For example, the values for energies below 0.5 MeV don't match the limit for 0.5 MeV (as they should per the Attachment 1 procedures). All the limits in the examples should be verified versus the figures since many appear to be incorrect.
Also, some DCF values appear to be high for some energy levels due to rounding up or due to that energy being between the energies shown in Figure A-2; the conservative approach would be to round down or use the lower DCF from the plotted energies. The procedures should be clarified to address these aspects too. For item e, staff noticed that the activity for energy group 4 in example 2 appeared to be lower than it is and energy group 8 in example 4 is missing the converted beta (to gamma) source from Y-90. The applicant should ensure the examples are correct since a package user may rely on them to guide its determination of acceptance of the contents in a given shipment. This information is required to demonstrate compliance with 10 CFR 71.81.

Response:
Attachment 1 has been completely rewritten and includes the requested revision.

7-5 Justify using the DCF in Figure A-2 for a payload density of 0.5 g/cc for payloads that have densities less than 0.5 g/cc.
Figure A-2 of the application does not include densities below 0.5 g/cc. However, the curve indicates the DCF value would drop significantly if it were extended to lower payload densities. Thus, it would be non-conservative to use the DCF at a density of 0.5 g/cc for payloads with lower densities. If payloads with lower densities are not shipped, then the procedures could be modified to limit allowable payloads to those with densities of 0.5 g/cc and above.
This information is required to demonstrate compliance with 10 CFR 71.47 and 71.87.

Response:
Attachment 1 has been completely rewritten and the new approach does not include the DCF.

7-6 Clarify the procedures for using the DCF for gamma energies not specifically plotted in Figure A-2 to indicate that the DCF for a specific gamma energy (or energy group) will be the smallest DCF of the nearest gamma energies (above and below) to the specific gamma energy (or energy group) being evaluated.
In response to the previous RAI 7-6, the applicant modified step 6 of Attachment A-1 to address this question. Specifically, the step now includes the direction that the "Applicable DCF is the smallest value at a particular density from the range of energy curves." However, the meaning of this statement is not clear. Further, it is not clear that the examples in the attachment are consistent with this statement.
The procedure should be modified to indicate that for a gamma energy (or energy group) that does not match those energies plotted in Figure A-2, the appropriate DCF is the lower of the DCF for the plotted gamma energy that is just above and the DCF for the plotted gamma energy that is just below the gamma energy (energy group) for which the DCF is being determined. The examples should also be modified to be consistent with this approach.
This information is required to demonstrate compliance with 10 CFR 71.47 and 71.87.

Response:
Attachment 1 has been completely rewritten and the new approach does not include the DCF.

Chapter 8 – Acceptance Tests and Maintenance Program

8-1 Provide a description of the acceptance leakage tests for packages fabricated after April 1, 1999, but before January 1, 2011.
In response to the previous RAI 8-5, the applicant modified its acceptance testing to address all package configurations to be used under the certificate of compliance. However, it appears that there are some packages for which an acceptance leakage test is not specified. Section 8.2 is for packages fabricated after April 1, 1999. However, the leakage tests in Section 8.2.4 are only for packages fabricated after January 1, 2011. Thus, the applicant needs to provide a description of the leakage tests that apply to packages fabricated between these two dates. If Section 8.2.4 is to apply to all packages fabricated after April 1, 1999, then the January 1, 2011, date should be deleted from the text in that section.
This information is required to demonstrate compliance with 10 CFR 71.85 and to ensure that the package performance will meet the requirements of 10 CFR Part 71, Subpart E.

Response:

Section 8.2.4 has been revised to delete the date; the section applies to all fabrication after April 1, 1999.

8-2 Modify the shielding acceptance tests to include more of the details regarding the acceptance test and criterion determination described in the response to the previous RAI 8-7 and to clarify how the acceptance criterion will be applied.
The response to the previous RAI 8-7 indicated the test will be performed with a gamma source positioned in the cask and that the acceptance criterion will be determined by calculation of the expected dose rates on the cask exterior. This information is an important aspect of the acceptance tests and should be included in the test description in the application. Additionally, the description of how the criterion is applied is not clear. It should indicate that packages for which any dose rate measurement indicates the lead is less than the minimum specified in the drawing in the current certificate of compliance shall not be acceptable and shall be remedied and retested.
This information is required to demonstrate compliance with 10 CFR 71.85(a).

Response:
The shielding acceptance tests for packages fabricated after April 1, 1999 (Step 8.2.6) has been modified as follows:

Shielding integrity of the package will be verified by gamma scan to assure the package lead layer meets or exceeds the minimum thickness specified on the cask drawing. All gamma scanning will be performed on a 4-inch square or less grid system. The acceptance criteria (maximum value) will be determined by: Option 1) measurement of the maximum value using a test block, which has shield layers that replicate the cask geometry per the drawing, using the gamma scan source and reproducing the source/shield/detector geometry that will be used during the scan of the cask, or Option 2) calculation of the maximum value using detailed modeling software (MCNP or equivalent) incorporating the specific cask dimensions from the drawing and the source/shield/detector geometry applicable to the gamma scan. Any location on the cask which shows a gamma scan value greater than the maximum value will be identified as unacceptable. All unacceptable areas will be remedied and re-scanned.

The shielding acceptance test for packages fabricated prior to April 1, 1999 (Step 8.1.5) has not been modified from the current SAR acceptance test.

8-3 Include a visual inspection of accessible surfaces of the packaging for significant defects, e.g., large dents, prior to loading and provide a statement indicating that significant damage to the packaging will preclude shipment until the cask repairs are made to bring the packaging into conformance with the licensing drawings.
The presence of large dents or defects to the packaging could influence the safety performance of the packaging during HAC. The current language in the application mandates only that EnergySolutions will be contacted if damage to the package is observed.
This information is required to demonstrate compliance with 10 CFR 71.33(a)(5) and 71.71(c)(4).

Response:
The following statement has been added to the note at step 7.1:

The cask may not be used as a Type B package until the damage is assessed by EnergySolutions and repairs, if required, are made to achieve conformance with the licensing drawings.

8-4 Justify that the foam density is consistent in the impact limiters or justify that the thermal and mechanical properties of the foam in the impact limiters will be bounded by significant deviations in foam density.
A recent, initial inspection finding discovered that Last-A-Foam used in a Part 71 transportation package had a density and crush strength more than 50% greater than what was required in the package's application. Yet, foam poured into shapes dissimilar to transportation package's impact limiters for acceptance testing met the density requirements listed in the application.
This information is required to demonstrate compliance with 10 CFR 71.33(a)(5), 71.71(c)(1), and 71.71(c)(4).

Response:
The foam procurement specification ES-M-175, Rev.1 specifies the stress-strain properties within an acceptable band. It has been demonstrated in Chapter 2 that within these tolerances the impact limiter provides acceptable protection to the cask during the HAC events. It has also been demonstrated that the average bulk properties of the foam material governs the energy absorption response of the impact limiter during various HAC events. Therefore, a local variation in density has a minor effect on the functionality of the impact limiters. Specification ES-M-175 requires the average density over the entire impact limiter to be within the acceptance band. The large variation in the crush strength identified by NRC staff may have been due to a complex shape of the impact limiter. The 8-120B impact limiter has very simple shape and uses a straight forward pouring process which will not result in a significant density and/or crush strength variation. Thus, Energy*Solutions* specifications are expected to result in uniform foam properties with only difference in parallel-to-rise and perpendicular-to-rise directions, which has been accounted for in the analyses of Chapter 2.

The foam thermal properties have not been used in the analyses of Chapter 3. Instead, the foam impact limiters are represented by appropriate thermal boundary conditions. Therefore, a variation in the thermal properties of the foam material will not have any effect on the analyses of Chapter 3.

1.0 General Information

1.1 Introduction

This Safety Analysis Report describes a reusable shipping package designed to protect radioactive material from both normal conditions of transport and hypothetical accident conditions. The package is designated the Model 8-120B package.

1.2 Package Description

1.2.1 Packaging

The package consists of a steel and lead cylindrical shipping cask with a pair of cylindrical foam-filled impact limiters installed on each end. The package configuration is shown in Figure 1.2-1. The internal cavity dimensions are 61 $\frac{13}{16}$ inches in diameter and 75 inches high. The cylindrical cask body is comprised of a 1½ inch thick external steel shell and a ¾ inch internal steel shell. The annular space between the shells is filled with 3.35 inch thick lead. The base of the cask consists of two 3¼ inch thick flat circular steel plates. The cask lid consists of two 3¼ inch thick flat circular steel plates. The lid is fastened to the cask body with twenty 2-8 UN bolts. There is a secondary lid in the middle of the primary lid. This secondary lid is attached to the primary lid with twelve 2-8 UN bolts.

The impact limiters are 102 inches in outside diameter and extend 22 inches beyond each end of the cask. There is a 50.0 inch diameter void at each end. Each impact limiter has an external shell, fabricated from ductile low carbon steel, which allows it to withstand large plastic deformations without fracturing. The volume inside the shell is filled with a crushable shock and thermal insulating polyurethane foam. The polyurethane is sprayed into the shell and allowed to expand until the void is completely filled. The foam bonds to the shell, which creates a unitized construction for the impact limiters.

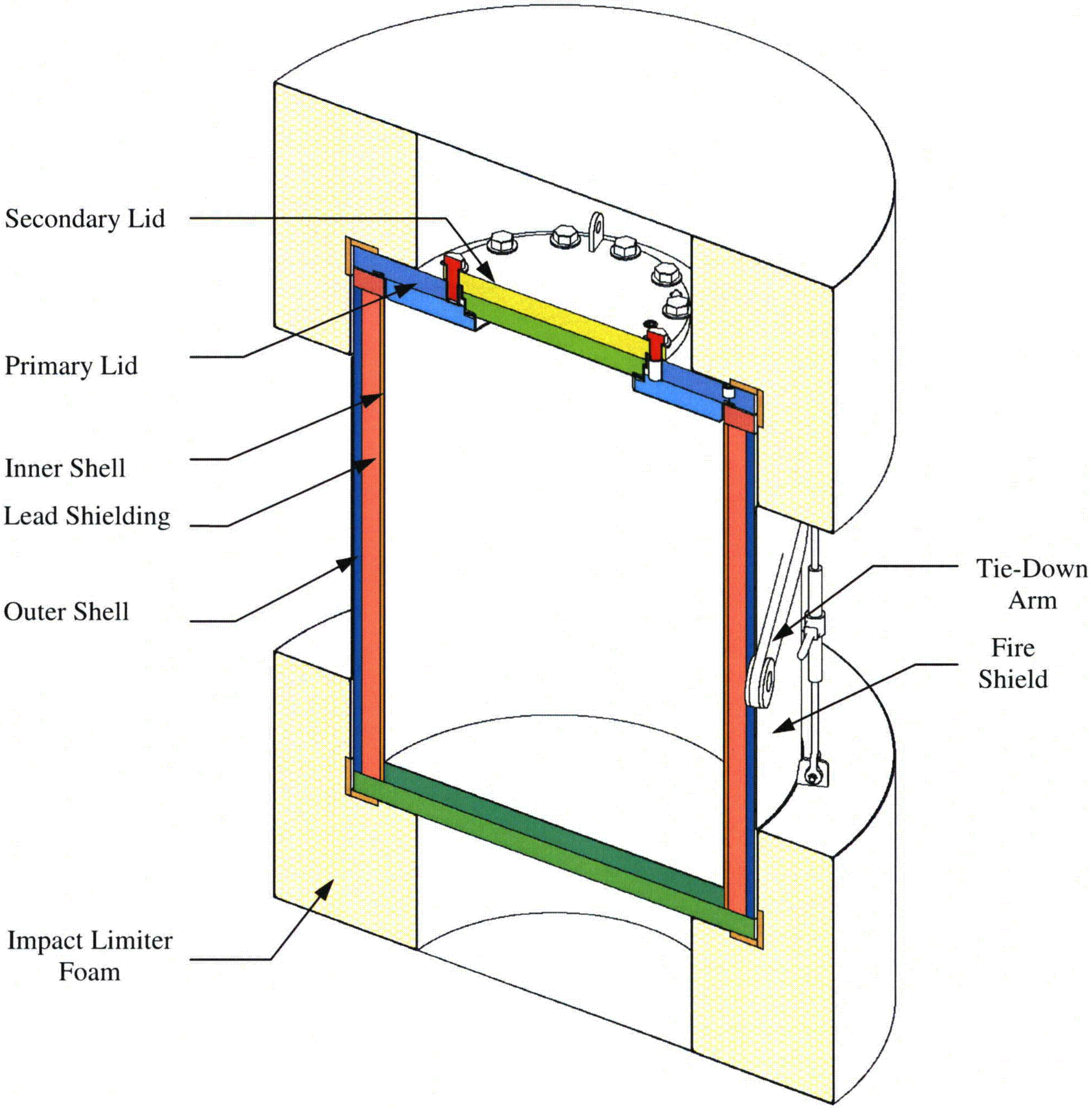


Figure 1-1
Features of the 8-120B Cask

The properties of the foam are further described in Section 2.2. The top and bottom impact limiters are connected together by eight one-inch diameter ratchet binders. This serves to hold the impact limiters in place on the cask during shipment, while allowing easy removal of the impact limiters for loading and unloading operations.

A general arrangement drawing of the package is included in Appendix 1.3. It shows the package dimensions as well as all materials of construction.

1.2.1.1 Containment Vessel

The containment vessel is defined as the inner steel shell of the cask body together with closure features comprised of the lower surface of the cask lid and 20 equally spaced 2-8 UN closure bolts.

1.2.1.2 Neutron Absorbers

There are no materials used as neutron absorbers or moderators in the package.

1.2.1.3 Package Weight

Maximum gross weight for the package is 74,000 lbs. including a maximum payload weight of 14,680 lbs.

1.2.1.4 Receptacles

There are no receptacles on this package.

1.2.1.5 Vent, and Test Ports

Pressure test ports with manual venting features exist between the twin o-ring seals for both the primary and secondary lids. This facilitates leak testing the package in accordance with ANSI N14.5.

The vent port is provided with the same venting features for venting pressures within the containment cavity, which may be generated during transport, prior to lid removal. Each port is sealed with an elastomer gasket. Specification information for all seals and gaskets is contained in Chapter 4.

1.2.1.6 Lifting Devices

Lifting devices are a structural part of the package. From the General Arrangement Drawing shown in Appendix 1.3, it can be seen that two removable lifting ears are provided, which attach to the cylindrical cask body. Three lifting lugs are also provided for removal and handling of the lid. Similarly, three lugs are provided for removal and handling of the secondary lid. Refer to Section 2.5.1 for a detailed analysis of the structural integrity of the lifting devices.

1.2.1.7 Tie-downs

From the General Arrangement Drawing, shown in Appendix 1.3, it can be seen that the tie-down arms are an integral part of the external cask shell. Consequently, tie-down arms are considered a structural part of the package. Refer to Section 2.5.2 for a detailed analysis of the structural integrity of the tie-down arms.

1.2.1.8 Heat Dissipation

There are no special devices used for the transfer or dissipation of heat.

1.2.1.9 Coolants

There are no coolants involved.

1.2.1.10 Protrusions

There are no outer or inner protrusions except for the tie-down arms described above. Lifting lugs are removed prior to transport.

1.2.1.11 Shielding

Cask walls provide a shield thickness of 3.35 inches of lead and 2¼ inches of steel. Cask ends provide a minimum of 6½ inches of steel. The contents will be limited such that the radiological shielding provided (4½ inches lead equivalent) will assure compliance with DOT and IAEA regulatory requirements.

1.2.1.12 Configurations

There are three configurations of the 8-120B cask.

- Configurations 1 and 2 were fabricated per the previously approved drawing Rev. 13 and differ mainly in the inclusion (Configuration 1) or lack (Configuration 2) of the optional drain port. Configuration 1 now includes sealing the drain port with the insertion and welding of a rod in the drain port. Acceptance Testing of Configurations 1 and 2 are described in Section 8.1. Fabrication of Configurations 1 or 2 after April 1, 1999 are not permitted.

- Configuration 3 does not have a drain port and the base plate is fabricated differently than Configurations 1 and 2. Acceptance Testing of Configuration 3 is described in Section 8.2.
- Configurations 1, 2 and 3 have the same Operations and Maintenance requirements and are described in Sections 7.0 and 8.3 respectively

All configurations have the same structural, thermal, containment, shielding, and criticality evaluations.

1.2.2 Contents of Packaging

1.2.2.1 Type form of material:

(1) Byproduct, source, or special nuclear material, in the form of dewatered resins, solids, including powdered or dispersible solids, or solidified material, contained within secondary container(s); or
(2) Radioactive material in the form of neutron activated metals or metal oxides in solid form contained within secondary container(s).

1.2.2.2 Maximum quantity of material per package:

Type B quantity of radioactive material not to exceed $3000A_2$, 200 thermal watts, and 14,680 pounds including weight of the contents, secondary container(s) and shoring. The contents may include fissile materials provided at least one of the paragraphs (a) through (f) of 10 CFR 71.15 is met. Materials producing neutrons through α, n or γ,n reactions or through spontaneous fission, other than fissile materials as allowed above, are not authorized.

The activity of beta and gamma emitting radionuclides shall not exceed the limit determined per the procedure in Chapter 7 Attachment 1.
Powdered or dispersible solid radioactive materials must have a mass of at least 60 grams or a specific activity of 50 A_2/g or less.

1.2.2.3 Loading Restrictions

Contents shall be packaged in secondary containers. Except for close fitting contents, shoring must be placed between the secondary containers or activated components and the cask cavity to prevent movement during accident conditions of transport.

Explosives, non-radioactive pyrophorics, and corrosives (pH less than 2 or greater than 12.5), are prohibited. Pyrophoric radionuclides may be present only in residual amounts less than 1 weight percent. Materials that may auto-ignite or change phase (i.e., change from solid to liquid or gas) at temperatures less than 350°F, not including water, shall not be included in the contents. In addition, as required by 10 CFR 71.43 (d), the contents shall not include any materials that may cause any significant chemical, galvanic, or other reaction.

Powdered solids shipments shall be performed only when the most recent periodic leak test meets the requirements of Chapter 4, Section 4.8. Powdered solid radioactive material shall not include radioactive forms of combustible metal hydrides, combustible elemental metals, i.e., magnesium, titanium, sodium, potassium, lithium, zirconium, hafnium, calcium, zinc, plutonium, uranium, and thorium, or combustible non-metals, i.e., phosphorus.

For any package containing water and/or organic substances which could radiolytically generate combustible gases, a determination must be made that , over a period of time that is twice the expected shipping time, the hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume (or equivalent limits for other inflammable gases) of the secondary container gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F).
The determination of hydrogen generation will be made using the methods in NUREG/CR-6673, *Hydrogen Generation in TRU Waste Transportation Packages*. NUREG/CR-6673 has equations that allow prediction of the hydrogen concentration as a function of time for simple nested enclosures and for packages containing multiple contents packaged within multiple nested confinement layers. The inputs to these equations include the bounding effective $G(H_2)$-value for the contents, the $G(H_2)$-values for the packaging material(s), the void volume in the containment vessel and in the confinement layers (when applicable), the temperature when the package was sealed, the temperature of the package during transport, and the contents decay heat.

For any package delivered to a carrier for transport, the secondary container must be prepared for shipment in the same manner in which the determination for gas generation is made. Shipment period begins when the package is prepared (sealed) and must be completed within twice the expected shipping time.

For any package containing materials with radioactivity concentration not exceeding that for LSA and shipped within 10 days of preparation, or within 10 days of venting the secondary container, the gas generation determination above need not be made and the shipping time restriction does not apply.

1.2.3 Special Requirements For Plutonium

Any contents that contain more than 0.74 TBq (20 Ci) of plutonium must be in solid form.

1.2.4 Operational Features

Refer to the General Arrangement Drawing of the package in Appendix 1.3. There are no complex operational requirements associated with the package

1.3 APPENDIX

CNS 8-120B Shipping Cask Drawing

Withheld from public disclosure as security-related sensitive information

4.0 Containment

This chapter describes the containment configuration of the Model CNS 8-120B Package for Normal Transport and Hypothetical Accident Conditions.

4.1 Description of Containment System

4.1.1 Containment Vessel

The package containment vessel is defined as the inner shell of the shielded transport cask, together with the associated lid, o-ring seals and lid closure bolts. The inner shell of the cask or containment vessel consists of a right circular cylinder of 62 inches inner diameter and 75 inches inside height. The shell is fabricated of ¾" thick carbon steel plate, ASTM A516-70. At the base, the cylindrical shell is attached to a circular end plate with full penetration welds. The primary lid is attached to the cask body with twenty (20) equally spaced 2-8 UN bolts. A secondary lid covers an opening in the primary lid and is attached to the primary lid using twelve (12) equally spaced 2-8 UN bolts. See Section 4.1.4 for closure details.

4.1.2 Containment Penetration

There are three penetrations of the containment vessel. These are (1) the primary lid with the containment boundary of the primary lid's inner o-ring; (2) the secondary lid with the containment boundary of the secondary lid's inner o-ring; and (3) the cask vent port located in the primary lid. A vent port penetrates the primary lid into the main cask cavity. The vent penetration is sealed with a Parker Stat-O-Seal. The primary and secondary lids are sealed with elastomeric o-rings.

4.1.3 Welds and Seals

The containment vessel is fabricated using full penetration groove welds. Seals (vent seal and o-rings) meet the requirements of ES Specification, ES-C-038 [Ref. 4.4].

4.1.4 Closure

The primary lid closure consists of two 3-1/4" thick laminated plates, stepped to fit over and within the top edge of the cylindrical body. The lid is supported at the perimeter of the cylindrical body by a thick plate (bolt ring) welded to the top of the inner and outer cylindrical body walls. This plate contains a 14-gauge stainless steel ring at a location, which corresponds to the sealing surface for the o-rings mounted in the lid. The lid is attached to the cask body by twenty (20) equally spaced 2-8 UN bolts. These bolts are torqued to 500 ft-lbs ± 10 % (lubricated). Two (2) solid elastomeric o-rings are retained in machined grooves at the lid perimeter. Groove dimensions prevent over-compression of the o-rings by the closure bolt pre-load forces and hypothetical accident impact forces. The cask is fitted with a secondary lid of similar construction attached to the primary lid with twelve (12) equally spaced identical bolts. The secondary lid is also sealed with two (2) solid, elastomeric o-rings in machined grooves.

The vent penetration is sealed with a Parker Stat-O-Seal, which is used beneath the heads of the hex head cap screws. Table 4.1 gives the torque values for the cap screws.

Location	Size (in.)	Torque Values (ft-lbs, ± 10% lubricated)
Vent Seal Bolt	1/2	20
Primary Lid	2-8UN	500
Secondary Lid	2-8UN	500

TABLE 4.1. Bolt and Cap Screw Torque Requirements

4.2 CONTAINMENT UNDER NORMAL CONDITIONS OF TRANSPORT

The 8-120B package is designed, fabricated, and leak tested to preclude release of radioactive materials in excess of the limits prescribed in 10CFR71.51(a)(1).

Of the permitted contents discussed in Section 1, two are considered in the following calculations as representative of the various types and forms permitted in the 8-120B; powdered solids and irradiated hardware. In this section and Section 4.2.1 below, the maximum permitted reference leakage rates (as defined in ANSI N14.5 – 1997 [Ref. 4.1]) for normal and hypothetical accident conditions are calculated for powdered solids and irradiated hardware waste forms, and the most restrictive of these (ie, the smallest leakage rate permitted) is taken as the reference leakage rate for the 8-120B cask and the basis for the acceptance criteria for leak testing. It is shown that the reference leakage rate (L_R) for the 8-120B cask is 1.54×10^{-6} ref-cm^3/sec, and that the release limits specified in 10CFR 71.51(a) (1) are met by limiting the release rate of the 8-120B to less than this value.

As discussed above, the most limiting type of radioactive waste contents permitted in the 8-120B is either powdered solids or irradiated hardware. The maximum permitted volumetric and reference leakage rates for Normal Conditions of Transport (NCT) are calculated for powdered solids and irradiated hardware ($L_{R\_N\_PS}$ and $L_{R\_N\_IH}$, respectively). Similar calculations are performed in Section 4.3 for Hypothetical Accident Conditions (HAC) ($L_{R\_A\_PS}$ and $L_{R\_A\_IH}$, respectively). The most restrictive of these four values is taken to be the maximum permitted reference leakage rate, L_R.

4.2.1 Maximum Permitted Leak Rate

In this section the maximum permitted leakage rate under Normal Conditions of Transport is calculated for the 8-120B package. 10CFR71.51(a)(1) states that the containment requirements for normal conditions of transport are:

> *...no loss or dispersal of radioactive contents, as demonstrated to a sensitivity of $10^{-6}A_2$ per hour, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the packaging.*

ANSI N14.5-1997 (Ref 4.1) states that the permissible leak rate shall be determined by Equation 4-1 below:

$$L := \frac{R}{C} \cdot \frac{cm^3}{sec} \qquad \text{Eqn. 4-1}$$

Where:

L = permissible volumetric leak rate (cm^3/sec)
R = package containment requirements (Ci/sec)
C = activity per unit volume of the medium that could escape from the containment system (Ci/cm^3)

For normal conditions of transport:

$$R_N := A_2 \cdot 10^{-6} \cdot \frac{1}{hr} \quad \Rightarrow \quad R_N = 2.78 \times 10^{-10} \frac{A_2}{sec} \qquad \text{10CFR71}$$

Determine the volume of the 8-120B cavity using dimensions from SAR drawing (Ref. 4.2):

$L_{cavity} := 75 \cdot in$ $\qquad L_{cavity} = 190.5 \cdot cm$

$D_{cavity} := 61.8 \cdot in$ $\qquad D_{cavity} = 156.972 \cdot cm$

The void volume of a typical hardware shipment and a powdered solids shipment are, respectively, 68% and 37% of the cask cavity volume. For leak rate calculations, the void volume (V_{cavity}) is conservatively assumed to be 25% of the cavity volume. Therefore,

$$V_{cavity} := \frac{(.25)\pi \cdot D_{cavity}^2 \cdot L_{cavity}}{4} \quad \Rightarrow \quad V_{cavity} = 9.217 \times 10^5 \cdot cm^3$$

In Sections 4.2.2 and 4.2.3 below, the maximum permitted volumetric leak rates under normal conditions of transport (L_N) are calculated for powdered solids and irradiated hardware respectively, and each is then converted into a reference leak rates (L_{R_N}).

4.2.2 Containment Under Normal Conditions of Transport (Powdered Solids)

Note: the following calculation for L_{N_PS} follows the methodology in NUREG/CR-6487 (Ref. 4.3)

C_{NPS} = concentration of releasable material during normal conditions of transport, C_i/cm^3

ρ = density of powder aerosol, g/cm^3

$\rho = 1 \times 10^{-6}$ g/cm^3 from NUREG/CR-6487 (Ref. 4.3)

Assume the mass (M) of the powdered solid is 60 grams and the activity (A) is 3000 A_2.

S_A = specific activity of the releasable material, A_2/g; = A/M = 50 A_2/g

$$C_{NP} := S_A \cdot \rho$$

Using Eqn. 4-1:

$$L_{N\_PS} := \frac{R_N}{C_{NPS}}$$

Then, $L_{N\_PS} = 5.556 \times 10^{-6}$ cm^3/sec **Maximum permitted volumetric leakage rate, normal conditions, powdered solids under the condition that the mass exceeds 60 grams or S_A is less than 50.**

Next, determine the Reference Leakage Rate, $L_{R\_N\_PS}$, normal conditions, powdered solids, for a volumetric leak rate L_{N_PS}:

$\mu_{air} := 0.0214 \cdot cP$ $\qquad M_{air} := 29.0 \cdot \frac{gm}{mole}$ $\qquad$ Ref. 4.1

$a := 0.6 \cdot cm$ $\qquad$ assumed length for hole leaking air (equals o-ring diameter)

For normal conditions of transport:

$T_N = 180$ deg F $\qquad$ MNOP = $P_{u\_N} = 35$ psig $\qquad$ from Chapter 3

$P_{u\_N} := 3.38$ atm

$P_{d\_N} := 1.0 \cdot atm$

$$P_{a\_N} := \frac{P_{u\_N} + P_{d\_N}}{2} \qquad P_{a\_N} = 2.19at$$

Use Eqn. B.3, B.4, and B.5 in ANSI N14.5 - 1997. Determine the diameter of a hole, D_{max1} that would leak L_{N_PS}.

$L_{N\_PS} := 5.56 \cdot 10^{-6} \cdot \frac{cm^3}{sec}$ $\qquad$ From above.

$$F_{mn}(D_{max}) := \frac{\left[3.8 \cdot 10^3 \cdot (D_{max} \cdot cm)^3 \cdot \sqrt{\frac{T_N \cdot gm}{M_{air} \cdot K \cdot mole}}\right] \cdot cm}{a \cdot P_{a\_N} \cdot sec}$$ Eqn B.3 from ANSI N14.5 - 1997

Also,

$$F_{cn}(D_{max}) := \frac{2.49 \cdot 10^6 \cdot (D_{max} \cdot cm)^4 \cdot cP}{a \cdot \mu_{air} \cdot atm \cdot sec}$$ Eqn B.4 from ANSI N14.5 - 1997

Use Eqn. B.5 from ANSI N14.5 - 1997. Let D_{max1} represent the diameter of the hole that will leak L_{N_PS}:

Solve for D_{max1}:

$$L(D_{max1}) := \left[\left(F_{cn}(D_{max1}) + F_{mn}(D_{max1})\right)\cdot\left(P_{u\_N} - P_{d\_N}\right)\cdot\frac{P_{a\_N}}{P_{u\_N}} \right] - L_{N\_PS}$$

$D_{max1} = 3.57 \times 10^{-4}$ cm

Now calculate $L_{R\_N\_PS}$ based on D_{max1}. At standard conditions:

$P_{u\_S} := 1.0\cdot atm$ $\qquad$ $P_{d\_S} := 0.1\cdot atm$

$P_{a\_S} = 0.55$ atm $\qquad$ $T_S := 298\cdot K$

Eqns B.3, B.4, and B.5 at standard conditions become:

$$F_{mstd}(D_{max}) := \frac{3.81\cdot 10^{3}\cdot(D_{max}\cdot cm)^{3}\cdot\sqrt{\frac{T_S\cdot gm}{M_{air}\cdot K\cdot mole}}\cdot cm}{a\cdot P_{a\_S}\cdot sec}$$

Simplify this equation:

$$F_{mstd}(D_{max}) \rightarrow 37010.092359370447894\cdot D_{max}{}^{3}\cdot\frac{cm^{3}}{atm\cdot sec}$$

$$F_{cstd}(D_{max}) := \frac{2.49\cdot 10^{6}\cdot D_{max}{}^{4}\cdot cm^{4}\cdot cP}{a\cdot \mu_{air}\cdot atm\cdot sec}$$

Simplify this equation:

$$F_{cstd}(D_{max}) \rightarrow 224324324.32432432432\cdot D_{max}{}^{4}\cdot\frac{cm^{3}}{atm\cdot sec}$$

Therefore, Eqn. B.5 at standard conditions and a hole diameter D_{max1} is:

$$L_{R\_N\_PS}(D_{max1}) := \left(F_{cstd}(D_{max1}) + F_{mstd}(D_{max1})\right)\cdot\left(P_{u\_S} - P_{d\_S}\right)\cdot\frac{P_{a\_S}}{P_{u\_S}}$$

Eqn B.5 from ANSIN14.5 - 1997

Thus,

$$L_{R\_N\_P}(D_{max1}) = 2.64\times 10^{-6}\frac{cm^{3}}{sec}$$

Standard leak rate, normal conditions, powdered solids.

4.2.3 Containment Under Normal Conditions of Transport (Irradiated Hardware)

Assume that the worst case source term for irradiated hardware is control rod blades having the same type and level of surface contamination as spent fuel, and that the potentially releasable contents from the control rod blades is entirely from this surface contamination. The surface contamination on the control rod blades that is equivalent to spent fuel is characterized in NUREG/CR-6487 (Ref. 4.3).

The following information was derived from Ref. 4.3, except as noted:

- bounding value for surface activity; worst case is for BWR fuel, $S_B = 1254 \times 10^{-6}$ Ci/cm^2
- surface area of control rod blade, $SA_B = 44{,}500$ cm^2, cruciform shape has 4 blade surfaces, blade width = 9.8", length conservatively assumed to be 175", A = 4 x 9.8" x 175", see Ref. 4.3
- A_2 for BWR fuel crud, normal transport conditions = 11.0 Ci
- fraction of surface activity that can spall off the surface of a blade and therefore is potentially releasable, normal transport conditions, $f_N = .15$

In addition, conservatively set the weight of control rod blade at 200 lbs, Ref. 4.3.

Given:

- weight capacity of 8-120B cask = 14680 lbs. (Chapter 1)
- number of control rod blades that can be transported in the 8-120B; assume 100% packing efficiency; N
- C_{NIH} = activity concentration in the cavity that could potentially escape during normal conditions of transport, irradiated hardware, C_i/cm^3
- total surface activity available for release on the surface of the control rod blades, normal transport conditions, RL_N:
- number of control rod blades in the cavity = N

N = 14680/200 N=73 blades

$$f_N := .15$$

$$S_B := 1254 \cdot 10^{-6} \cdot \frac{C_i}{cm^2}$$

$$SA_B := 44500 \cdot cm^2$$

$$RL_N := N \cdot S_B \cdot SA_B \cdot f_N \quad \Rightarrow \quad RL_N = 6.11 \times 10^2 \; C_i$$

$$C_{NIH} := \frac{RL_N}{V_{cavity} \cdot (11.0)} \quad \Rightarrow \quad C_{NIH} = 6.027 \times 10^{-5} \cdot cm^{-3}$$

from Eqn. 1-1 above:

$L_{N\_IH} := \frac{R_N}{C_{NIH}}$ $\quad L_{N\_I} = 4.609 \times 10^{-6} \frac{cm^3}{sec}$ **Maximum permitted volumetric leakage rate, normal conditions of transport, for irradiated hardware.**

Next, determine the Reference Leakage Rate, $L_{R\_N\_IH}$, normal conditions, irradiated hardware, for a volumetric leak rate L_{N_IH}:

Follow the same steps used above. First, determine a D_{max2} that would leak L_{N_IH}:

Use Eqn. 4-2:

$$L(D_{max2}) := \left[(F_{cn}(D_{max2}) + F_{mn}(D_{max2})) \cdot (P_{u\_N} - P_{d\_N}) \cdot \frac{P_{a\_N}}{P_{u\_N}} \right] - L_{N\_IH}$$

Solve this equation for D_{max2}:

$$D_{max2} := 3.4 \cdot 10^{-4} \, cm$$

Now substitute D_{max2} into Eqn. B.5 and determine $L_{R\_N\_IH}$ at standard conditions:

$$L_{R\_N\_IH}(D_{max2}) := \left(F_{cstd}(D_{max2}) + F_{mstd}(D_{max2})\right)\cdot\left(P_{u\_S} - P_{d\_S}\right)\cdot\frac{P_{a\_S}}{P_{u\_S}}$$

$L_{R\_N\_IH}(D_{max2}) = 2.20 \times 10^{-6}\ cm^3/sec$ **Standard leak rate, normal conditions, irradiated hardware.**

4.3 CONTAINMENT UNDER HYPOTHETICAL ACCIDENT CONDITIONS OF TRANSPORT (TYPE B PACKAGES)

In this section the maximum permitted leakage rates under Hypothetical Accident Conditions are calculated for the 8-120B package. 10CFR71.51(a)(2) states that the containment requirements for Hypothetical Accident Conditions are:

> *...no escape of krypton-85 exceeding 10A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package.*

Following the methodology from Section 4.2 in Sections 4.3.1 and 4.3.2 below, the maximum permitted volumetric leakage rates under Hypothetical Accident Conditions are calculated for powdered solids and irradiated hardware, L_{A_PS} and L_{A_IH} respectively. In Section 4.3.1 the reference leakage rate corresponding to L_{A_PS}, $L_{R\_A\_PS}$, is calculated, and in Section 4.3.2 the reference leakage rate corresponding L_{A_IH}, $L_{R\_A\_IH}$, is calculated.

In Section 4.4, $L_{R\_A\_PS}$ and $L_{R\_A\_IH}$ are compared to the reference leakage rates for Normal Conditions of Transport calculated in Section 4.2.1 to determine the most restrictive, and thus the reference air leakage rate for the 8-120.

$R_A := 1\cdot\frac{A_2}{week}$ $\qquad R_A = 1.65 \times 10^{-6}\frac{A_2}{sec}$ $\qquad$ 10CFR71

4.3.1 Containment Under Hypothetical Accident Conditions (Powdered Solids)

Use the same parameters as Section 4.2.2:

C_{APS} = concentration of releasable materials during hypothetical accident conditions, C_i/cm^3

$C_{APS} := C_{NPS}$

Using Eqn 1-1:

$$L_{A\_PS} := \frac{R_A}{C_{APS}}$$

$L_{A\_PS} = 0.033\ cm^3/sec$ **Volumetric leakage rate, hypothetical accident conditions, powdered solids**

Next, determine the reference leakage rate, $L_{R\_A\_PS}$, accident conditions, powdered solids, for a volumetric leak rate L_{A_PS}:

$P_{d\_A} := 1 \cdot atm$ $\mu_{air} := 0.0185 \cdot cP$ $M_{air} := 29.0 \cdot \frac{gm}{mole}$ Ref. 4.1

$a := 0.6 \cdot cm$ assumed length for hole leaking air (equals o-ring diameter)

For hypothetical accident conditions:

$T_A = 325$ deg F HACP = $P_{u\_A} = 155$ psig

$P_{u\_A} := 155$ psig From Section 3

$$P_{u\_A}(x) := (x \cdot psig + 14.7) \cdot psi$$

$$P_{u\_A} := 11.6 \cdot atm$$

$$P_{d\_A} := 1 \cdot atm$$

$$P_{a\_A} := \frac{P_{u\_A} + P_{d\_A}}{2} \qquad P_{a\_A} = 6.28 \text{ atm}$$

Equations B.3 and B.4 at accident conditions are as follows:

$$F_{mA}(D_{max}) := \frac{3.8 \cdot 10^3 \cdot (D_{max} \cdot cm)^3 \cdot \sqrt{\frac{T_A \cdot gm}{M_{air} \cdot K \cdot mole}} \cdot cm}{a \cdot P_{a\_A} \cdot sec}$$

Eqn B.3 from ANSI N14.5 - 1997

$$F_{mA}(D_{max}) \rightarrow \frac{3913.1984257554438542 \cdot D_{max}{}^3 \cdot cm^3}{atm \cdot sec}$$

$$F_{cA}(D_{max}) := \frac{2.49 \cdot 10^6 \cdot (D_{max} \cdot cm)^4 \cdot cP}{a \cdot \mu_{air} \cdot atm \cdot sec} \qquad F_{cA}(D_{max}) \rightarrow \frac{1.8526785714285714286e8 \cdot D_{max}{}^4 \cdot cm^3}{atm \cdot sec}$$

Eqn B.4 from ANSI N14.5 - 1997

Let D_{max3} represent the diameter of the hole that will leak L_{A_PS}:

$$L_{A\_PS} := 0.033 \cdot \frac{cm^3}{sec}$$

$$L(D_{max3}) := \left[\left(F_{cA}(D_{max3}) + F_{mA}(D_{max3})\right) \cdot \left(P_{u\_A} - P_{d\_A}\right) \cdot \frac{P_{a\_A}}{P_{u\_A}} \right] - L_{A\_PS}$$

Solve this equation for D_{max3}:

$$D_{max3} := 2.4 \cdot 10^{-3} \ cm$$

Substitute this value of D_{max3} into Eqn B.3 at standard conditions:

$$L_{R\_A\_PS}(D_{max3}) := \left(F_{cstd}(D_{max3}) + F_{mstd}(D_{max3})\right) \cdot \left(P_{u\_S} - P_{d\_S}\right) \cdot \frac{P_{a\_S}}{P_{u\_S}}$$

$$L_{R\_A\_PS}(D_{max3}) = 0.004 \frac{cm^3}{sec}$$ **Standard leak rate, accident conditions, powered solids.**

4.3.2 Containment Under Hypothetical Accident Conditions (Irradiated Hardware)

(See Section 4.4 for the basic assumptions regarding control rod blades and irradiated hardware.)
For accident conditions:

- A_2 for BWR fuel, accident conditions = 11.0 Ci (Ref. 4.3)
- fA = 1.0 (Ref. 4.3) fraction of surface activity potentially that can spall off surface of a blade and therefore is potentially releasable under accident conditions,

C_{AIH} = activity concentration in the cavity that could potentially escape during accident conditions, irradiated hardware, C_i/cm^3

$RL_A := N \cdot S_B \cdot SA_B \cdot f_A \quad RL_A = 4.07 \times 10^3 \cdot C_i$

$$C_{AIH} := \frac{RL_A}{V_{cavity} \cdot (11.0)} \qquad C_{AIH} = 4.02 \times 10^{-4} \frac{A_2 \cdot C_i}{c^3}$$

$$L_{A\_IH} := \frac{R_A}{C_{AIH}} \quad L_{A\_IH} = 4.12 \times 10^{-3} \cdot \frac{cm^3}{sec}$$

Volumetric leak rate, Hypothetical Accident Conditions, Irradiated hardware

Next, determine the reference leakage rate, $L_{R\_A\_IH,}$ accident conditions, irradiated hardware, for a volumetric leak rate L_{A_IH}:

Follow the same steps used in Section 4.3.1 above. First, determine a D_{max4} that would leak L_{A_IH}:

$L_{A\_IH} := 4.12 \cdot 10^{-3} \cdot \frac{cm^3}{sec}$ From above.

$$L(D_{max4}) := \left[\left(F_{cA}(D_{max4}) + F_{mA}(D_{max4})\right) \cdot (P_{u\_A} - P_{d\_A}) \cdot \frac{P_{a\_A}}{P_{u\_A}} \right] - L_{A\_IH}$$

Solve this equation for D_{max4}

$D_{max4} := 1.40 \cdot 10^{-3}$ cm

Now substitute D_{max4} into Eqn B.5 and determine $L_{R\_A\_IH}$ at standard conditions:

$$L_{R\_A\_IH}(D_{max4}) := \left(F_{cstd}(D_{max4}) + F_{mstd}(D_{max4})\right) \cdot (P_{u\_S} - P_{d\_S}) \cdot \frac{P_{a\_S}}{P_{u\_S}}$$

$$L_{R\_A\_IH}(D_{max4}) = 4.77 \times 10^{-4} \cdot \frac{cm^3}{sec}$$

Standard leak rate, accident conditions, irradiated hardware.

4.4 Reference Air Leakage Rate

The following table summarizes results in Sections 4.2 and 4.3 above:

	Max. Volumetric Leak Rate (cm^3/sec)	Max. Hole Diameter (cm)	Reference Leak Rate (cm^3/sec)
Normal Conditions of Transport, Powdered Solids	$L_{N\_PS} = 5.56 \times 10^{-6}$	$D_{max1} = 3.57 \times 10^{-4}$	$L_{R\_N\_PS} = 2.64 \times 10^{-6}$
Normal Conditions of Transport, Irradiated Hardware	$L_{N\_IH} = 4.61 \times 10^{-6}$	$D_{max2} = 3.4 \times 10^{-4}$	$L_{R\_N\_IH} = 2.20 \times 10^{-6}$
Hypothetical Accident Conditions, Powdered Solids	$L_{A\_PS} = 0.033$	$D_{max3} = 2.40 \times 10^{-3}$	$L_{R\_A\_PS} = 0.004$
Hypothetical Accident Conditions, Irradiated Hardware	$L_{A\_IH} = 4.12 \times 10^{-3}$	$D_{max4} = 1.40 \times 10^{-3}$	$L_{R\_A\_IH} = 4.77 \times 10^{-4}$

The reference leak rate for powdered solids was determined based on the assumption that the powdered solid source has a mass of at least 60 grams or the SA is less than 50. With these constraints, $L_{R\_N\_PS}$ is not the most restrictive leak rate. The most restrictive reference leak rate is $L_{R\_N\_IH}$, for normal conditions of transport, irradiated hardware, and will be the reference leak rate for the cask. Therefore, for the 8-120B cask:

$$L_R := 2.20 \cdot 10^{-6} \cdot \frac{ref \cdot cm^3}{sec}$$ **8-120B cask reference air leakage rate**

4.5 Determination of Equivalent Reference Leakage Rate for R-134a Gas

The purpose of this calculation is to determine the allowable leak rate using the R-134a halogen gas that may be used to perform the annual verification leak tests on the 8-120B cask. This halogen gas is now in widespread use as a replacement gas for R-12 in many industrial applications.

This calculation uses formulas presented in ANSI N14.5 - 1997.

$$L_R = 2.2 \times 10^{-6} \frac{cm^3}{sec}$$

As calculated above, maximum diameter hole through the O-ring corresponding to this leakage rate is:

$$D_{MAX} := D_{max2} \cdot cm \qquad \Rightarrow \qquad D_{MAX} = 3.4 \times 10^{-4} \cdot cm$$

Determine the equivalent air/R134a mixture (L_{mix}) that would leak from D_{MAX} during a leak test. Assume the cask void is first evacuated to 20" Hg vacuum (9.92" Hg abs) and then pressurized to 25 psig (2.7 atm) with an air/R134a mixture.

$P_{mix} := 2.701 \cdot atm \qquad P_{air} := 9.92 \cdot in\_Hg = 0.332 \cdot atm$

$P_{R134a} := P_{mix} - P_{air} \qquad \Rightarrow \qquad P_{R134a} = 2.37 \cdot atm$

$P_d := 1.0 \cdot atm$

$P_a := \dfrac{P_{mix} + P_d}{2} \qquad \Rightarrow \qquad P_a = 1.85 \cdot atm$

The properties of R134a are :

$M_{R134a} := 102 \cdot \dfrac{gm}{mole}$

$\mu_{R134a} := 0.012 \cdot cP$

$M_{mix} := \dfrac{M_{R134a} \cdot P_{R134a} + M_{air} \cdot P_{air}}{P_{mix}} \qquad M_{mix} = 93.04 \cdot \dfrac{gm}{mole}$ Eqn. B7 - ANSI N14.5

$\mu_{mix} := \dfrac{\mu_{air} \cdot P_{air} + \mu_{R134a} \cdot P_{R134a}}{P_{mix}} \qquad \mu_{mix} = 0.0128 \cdot cP$ Eqn. B8 - ANSI N14.5

Determine L_{mix} as a function of temperature. Assume the viscosities of air and R134a do not change significantly over the range of temperatures evaluated:

$T := 273, 278 .. 328$ °K Temperature range for test: 32°F to 130°F

$F_c := \dfrac{2.49 \cdot 10^6 \cdot D_{MAX}^4 \cdot cP \cdot ref}{a \cdot \mu_{mix} \cdot sec \cdot atm}$ then, $F_c = 4.541 \times 10^{-6} \cdot \dfrac{cm^3}{atm \cdot sec}$

$$F_m(T) := \frac{3.81 \cdot 10^3 \cdot D_{MAX}^3 \cdot \sqrt{\dfrac{T}{M_{mix}}} \cdot cm \cdot gm^{0.5}}{a \cdot P_a \cdot mole^{0.5} \cdot sec}$$

$$L_{mix}(T) := \left(F_c + F_m(T)\right)\cdot\left(P_{mix} - P_d\right)\cdot\frac{P_a}{P_{mix}}$$

$$T_F(T) := \left[(T - 273)\cdot\frac{9}{5} + 32\right]\cdot F$$

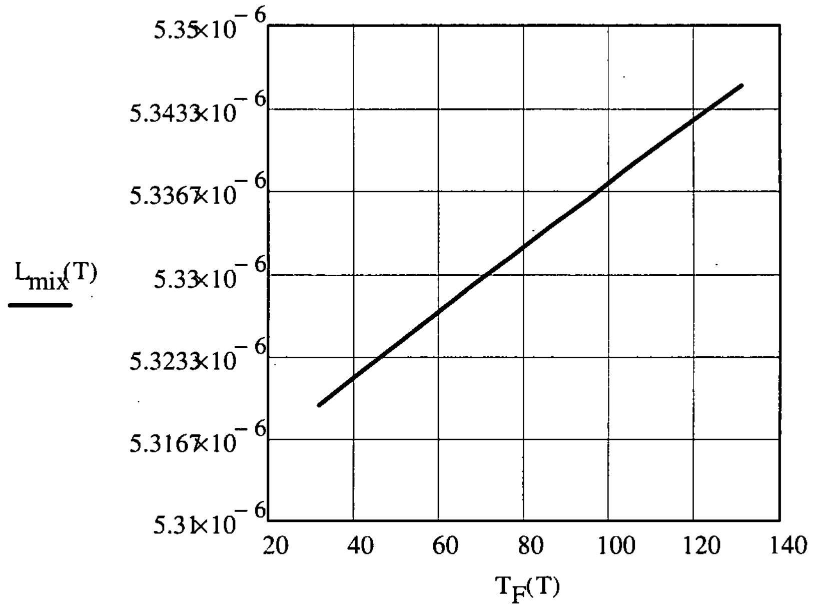


Fig.4.1 - Allowable R-134a/Air Mixture Test Leakage, cm^3/sec, versus test temperature, deg.F

The R-134a component of this leak rate can be determined by multiplying the leak rate of the mixture by the ratio of the R-134a partial pressure to the total pressure of the mix, as follows.

$$L_{R134a}(T) := L_{mix}(T)\cdot\frac{P_{R134a}}{P_{mix}}$$

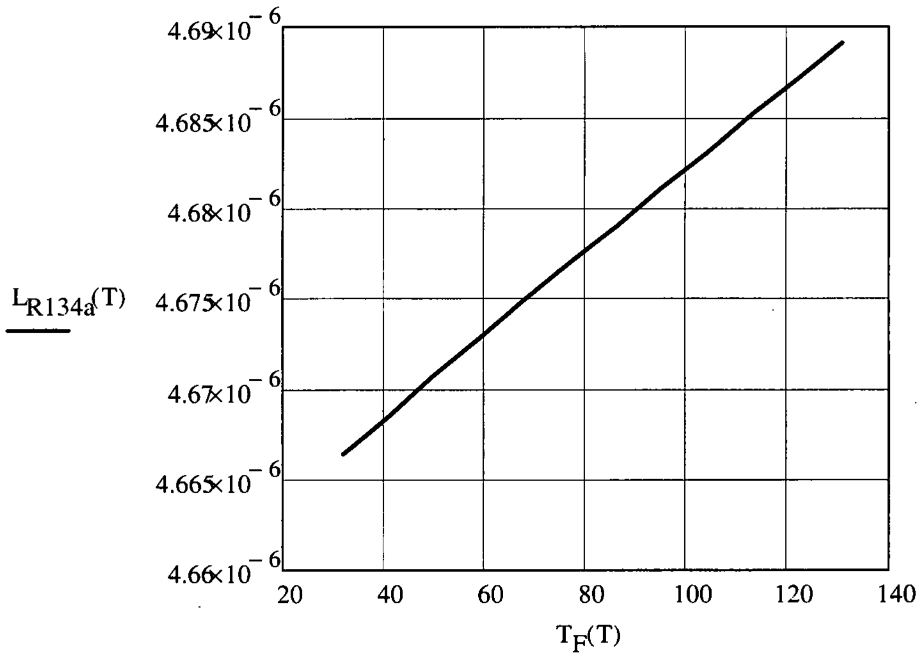


Fig. 4.2 - Allowable R-134a Test Leakage, cm^3/sec, versus Test Temperature, °F

Determine the equivalent mass flow rate for L_{R134a} in oz/yr:

$$N(T) := \frac{P_{R134a} \cdot V}{R_o \cdot T}$$ Ideal Gas Law

where,

$$R_o := \frac{82.05 \cdot cm^3 \cdot atm}{mole} \qquad V := 1 \cdot cm^3$$

This data can then be used to convert the volumetric leak rate for R-134a calculated above to a mass leak rate. By dividing N by V, the number of moles per unit volume can be multiplied by the molecular weight of the gas and the maximum allowable volumetric leak rate to determine the maximum allowable mass leak rate, as a function of test temperature as shown in the graph below. The conversion from grams per second to ounces per year is also shown below.

$$\frac{gm}{sec} = 1.11 \times 10^6 \frac{oz}{yr}$$ Conversion of gm/sec to oz/yr

$$L(T) := L_{R134a}(T) \cdot \frac{N(T)}{V} \cdot M_{R134a}$$

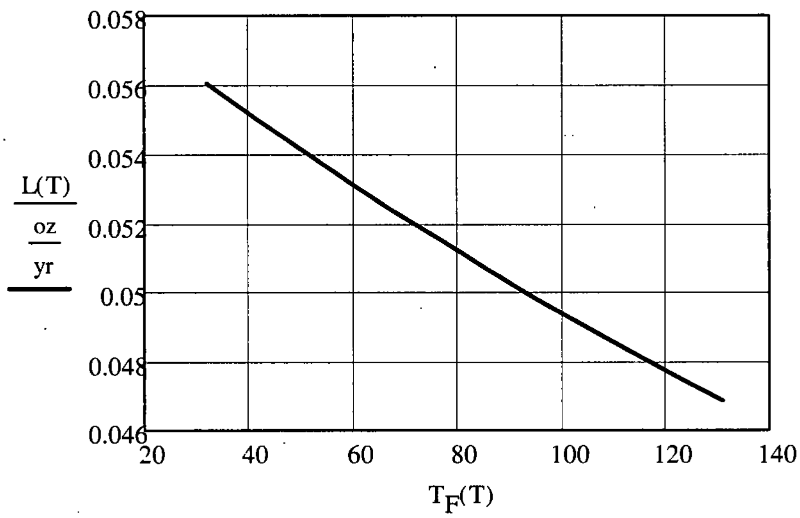


Fig. 4.3 - Allowable R-134a Test Leakage, oz/yr, versus Test Temperature, °F

Figure 4.4 can be used to determine the allowable leak rate based on the temperature at the time of the test. A simplified version of the equation can be used to validate the curve:

$L(T_F) = 4.872 \times 10^{-2} \times (5/9 \times T_F + 255.2)^{-0.5} + 15.28 \times (5/9 \times T_F + 255.2)^{-1}$

According to ANSI N14.5 methodology, the maximum allowable leak rate must be divided by 2 to determine the minimum sensitivity for the test. A graph of the required sensitivity in oz/yr is presented below:

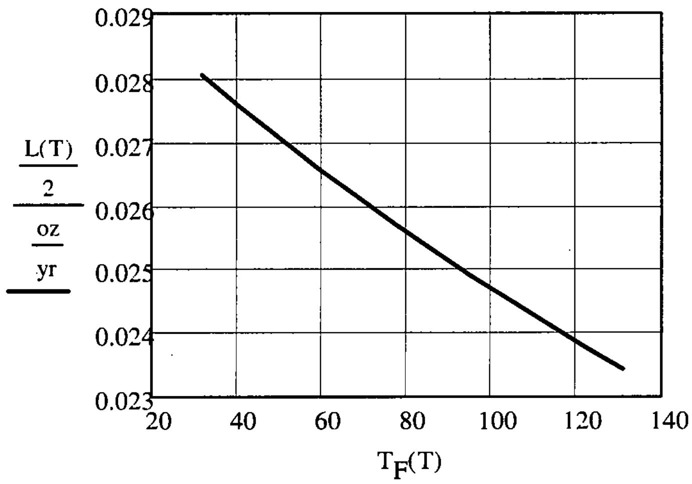


Fig. 4.4 - Allowable R-134a test leakage sensitivity, oz/yr, versus test temperature, °F

A simplified version of the equation can be used to validate the sensitivity curve:

$L(T_F)/2 = 2.436 \times 10^{-2} \times (5/9 \times T_F + 255.2)^{-0.5} + 7.64 \times (5/9 \times T_F + 255.2)^{-1}$

4.6 Determination of Equivalent Reference Leakage Rate for Helium Gas

The purpose of this calculation is to determine the allowable leak rate using the Helium gas that may be used to perform the annual verification leak tests on the 8-120B cask.

This calculation uses formulas presented in ANSI N14.5 - 1997.

$$L_R = 2.2 \times 10^{-6} \frac{cm^3}{sec}$$

As calculated above, maximum diameter hole through the O-ring corresponding to this leakage rate is:

$$D_{MAX} := D_{max2} \cdot cm \qquad \Rightarrow \qquad D_{MAX} = 3.4 \times 10^{-4} \cdot cm$$

Determine the equivalent air/helium mixture (L_{mix}) that would leak from D_{MAX} during a leak test. Assume the cask void is first evacuated to 20" Hg vacuum (9.92" Hg abs) and then pressurized to 1 psig (1.07 atm) with an air/helium mixture.

$P_{air} = 0.33\,atm$ $\quad$ $P_d = 0.01\,at$ $\quad$ $P_{mix} = 1.07\,at$

$$P_{He} := P_{mix} - P_{air}$$

$$P_a := \frac{P_{mix} + P_d}{2} \qquad P_a = 0.54\,atm$$

$$M_{He} := 4.0 \cdot \frac{g}{mol}$$ ANSI N14.5 - 1997

$$\mu_{He} := 0.0198 \cdot cP$$ ANSI N14.5 - 1997

$$M_{mi} := \frac{M_{He} \cdot P_{He} + M_{air} \cdot P_{air}}{P_{mi}} \qquad M_{mix} = 11.75\,\frac{gm}{mole}$$ Eqn. B7 - ANSI N14.5

$$\mu_{mi} := \frac{\mu_{air} \cdot P_{air} + \mu_{He} \cdot P_{He}}{P_{mi}} \qquad \mu_{mi} = 0.019cP$$ Eqn. B8 - ANSI N14.5

Determine L_{mix} as a function of temperature. Assume the viscosities of air and Helium do not change significantly over the range of temperatures evaluated:

$T := 273, 278 .. 328 \ {}^{\circ}K$ $\quad$ Temperature range for test: 32°F to 130°F

$$F_c := \frac{2.49 \cdot 10^6 \cdot D_{MAX}{}^4 \cdot cP \cdot std}{a \cdot \mu_{mix} \cdot sec \cdot atm}$$ Eqn. B3 – ANSI N14.5

$$F_m(T) := \frac{3.81 \cdot 10^3 \cdot D_{MAX}{}^3 \cdot \sqrt{\frac{T}{M_{mix}}} \cdot cm \cdot gm^{0.5}}{a \cdot P_a \cdot mole^{0.5} \cdot sec}$$ Eqn. B4 – ANSI 14.5

$$L_{mix}(T) := \left(F_c + F_m(T)\right) \cdot \left(P_{mix} - P_d\right) \cdot \frac{P_a}{P_{mix}}$$ Equation B5, ANSI N14.5

$$T_F(T) := \left[(T - 273)\cdot\frac{9}{5} + 32\right]\cdot F$$

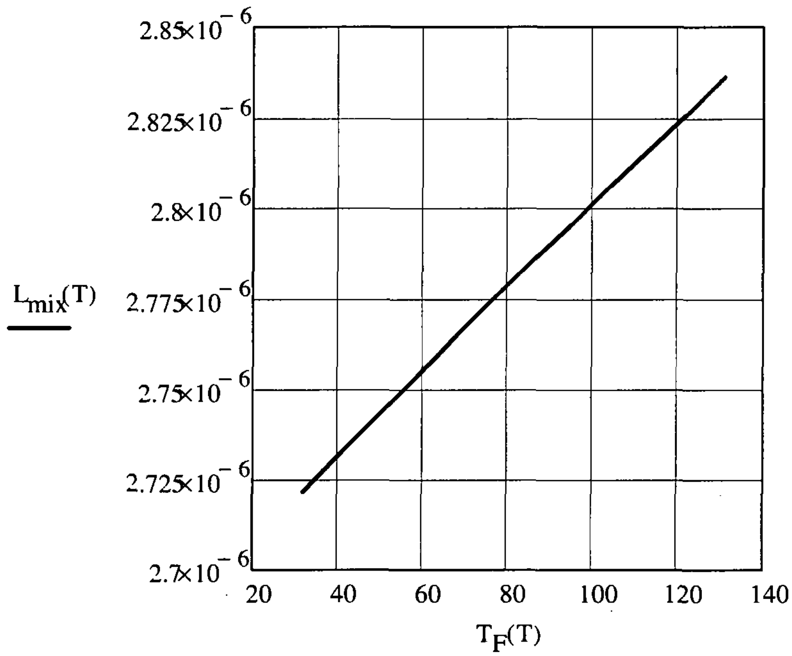


Fig. 4.5 - Allowable He/Air Mixture Test Leakage, cm^3/sec, versus test temperature, °F

The Helium component of this leak rate can be determined by multiplying the leak rate of the mixture by the ratio of the Helium partial pressure to the total pressure of the mix, as follows.

$$L_{He}(T) := L_{mix}(T)\cdot\frac{P_{He}}{P_{mix}}$$

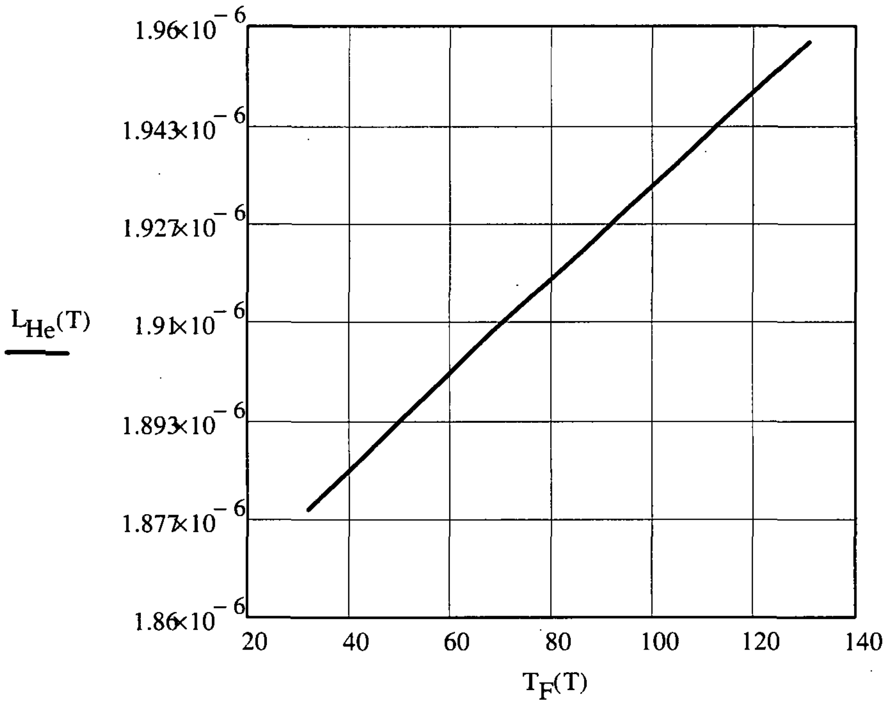


Fig. 4.6 - Allowable He Test Leakage, cm^3/sec, versus test temperature, °F

Figure 4.6 can be used to determine the allowable leak rate based on the temperature at the time of the test. A simplified version of the equation can be used to validate the curve:

$$L_{He}(T_F) = 2.114 \times 10^{-6} + 5.193 \times 10^{-8} \times (5/9 \times T_F + 255.2)^{0.5}$$

According to ANSI N14.1 methodology, the maximum allowable leak rate must be divided by 2 to determine the minimum sensitivity for the test. A graph of the required sensitivity is presented below.

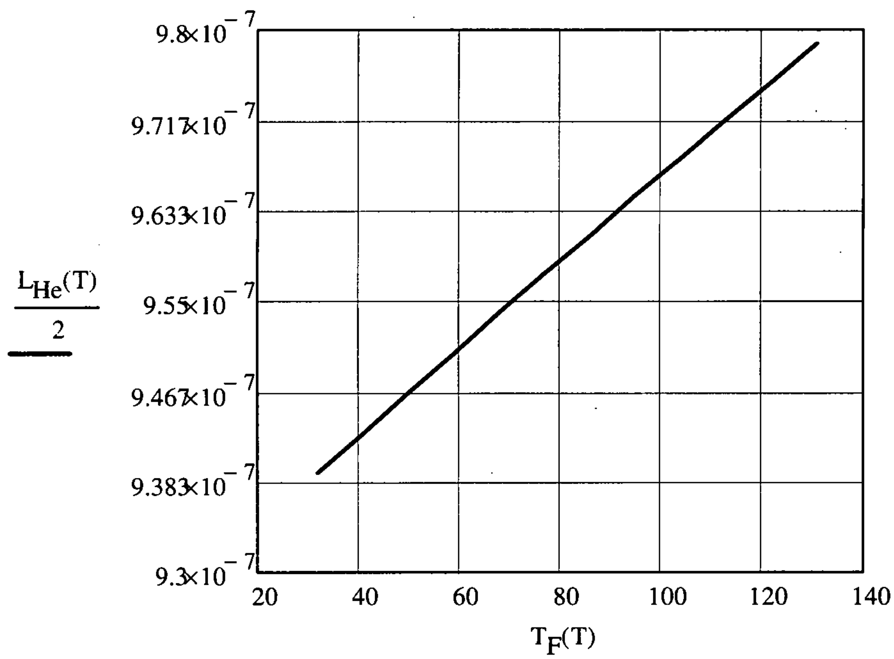


Fig. 4.7 - Allowable helium test leakage sensitivity, cm^3/sec, versus test temperature, °F

A simplified version of the equation can be used to validate the sensitivity curve:

$$L_{He}(T_F) = (2.114 \times 10^{-6} + 5.193 \times 10^{-8} \times (5/9 \times T_F + 255.2)^{0.5}) \div 2$$

4.7 Determining Time for Pre-Shipment Leak Test Using Air or Nitrogen

The pre-shipment leak test is to be performed by the pressure drop test method using air or nitrogen. The test will be performed on the closure lid, and may also be performed on the vent port if this has been operated since the last test. In this section the minimum hold time for each of the tests is determined.

4.7.1 Minimum Hold Time for Closure Lid

The pre-shipment leak test is performed by charging the annulus between the O-rings of the closure lid with air at 18 psig and holding the pressure for the prescribed time. The maximum volume of the test manifold is 10 cm^3, which is added to the annulus volume.
The annulus between the O-rings is 1/8" deep and 1/8" wide with a center-line diameter (primary lid) of 63 $^7/_8$". The volume of the annulus is:

$$ID_{ann} := \left[(63.875) - \frac{1}{8}\right]\cdot in \quad \Rightarrow \quad ID_{ann} = 63.75\cdot in$$

$$OD_{ann} := \left(63.875 + \frac{1}{8}\right)\cdot in \quad \Rightarrow \quad OD_{ann} = 64.00\cdot in$$

$$V_{ann} := \frac{\pi}{4}(.125\cdot in)\left(OD_{ann}{}^2 - ID_{ann}{}^2\right)$$

$$V_{ann} = 3.14\cdot in^3 \quad \Rightarrow \quad V_{ann} = 51.38\cdot cm^3$$

$$V_T := V_{ann} + 10\ cm^3 \qquad V_T = 61.4\ cm^3$$

Use Equation B.14 from ANSI N14.5 to determine the required hold time given the maximum permitted leak rate, where:

L = atm-cm^3 of air at standard conditions
V_{ann} = gas volume in the test annulus
T_s = reference absolute temperature, 298°K
H = test duration, hrs
P_s = standard pressure, 1 atm
P_1 = gas pressure in annulus at start of test, 1.232 atm (18.1 psig)
P_2 = gas pressure in annulus at end of test, 1.225 atm (18.0 psig)
T_1 = gas temperature in annulus at start of test, °K
T_2 = gas temperature in annulus at end of test, °K

$T_s := 298 \cdot K \qquad T_1 := T_s \qquad T_2 := T_s$

$P_s := 1 \cdot atm$

P_1 - P_2= P_{delta} Maximum permitted P_d = sensitivity of pressure gage:

$P_{delta} := .1 \cdot psi \qquad P_{delta} = 0.007 \cdot atm$

$$L := \frac{V_T \cdot T_s}{3600 \cdot H \cdot P_s} \cdot \left(\frac{P_1}{T_1} - \frac{P_2}{T_2} \right) \cdot \frac{cm^3}{sec}$$ **Eqn. 4.7-1**

The maximum permitted sensitivity for the pre-shipment leak test as prescribed in ANSI N14.5 - 1997 is 10^{-3} ref-cm^3/sec. From Equation B.17 in ANSI N14.5, the maximum permitted leak rate when the sensitivity is prescribed is:

$L \le S/2$ therefore,

$$L := \frac{10^{-3}}{2} \cdot \frac{cm^3}{sec}$$

Rearrange Eqn 4.7-1 to solve for H:

$$H := \frac{V_T \cdot T_s \cdot P_{delta}}{3600 \cdot \frac{sec}{hr} \cdot L \cdot P_s \cdot T_s}$$ **Eqn. 4.7-2**

$H = 13.92 \cdot min$

For conservatism, the test will be conducted for 15 minutes.

The smaller diameter secondary lid will be conservatively tested for the same time as the primary.

4.7.2 Minimum Hold Time for Vent Port

Volume of vent port cavity:

$$V_{vent} := \frac{\pi}{4}(1.875 \cdot in)^2 \cdot 1.125 \cdot in$$

Volume of seal plug head inside vent port cavity:

$$V_{seal} := \frac{\pi}{4}(1.5 \cdot in)^2 \cdot (1 \cdot in)$$

$$V_{test} := V_{vent} - V_{seal} \qquad V_{test} = 21.945 \cdot cm^3$$

$$V_T := V_{test} + 31.6 \cdot cm^3$$

$$H := \frac{V_T \cdot T_s \cdot P_{delta}}{3600 \cdot \frac{sec}{hr} \cdot L \cdot P_s \cdot T_s} \qquad H = 0.202 \cdot hr \qquad H = 12.145 \cdot min$$

For conservatism, the test will be conducted for 15 minutes.

4.8 Leakage Rate Tests for Type B Packages

The following leakage tests are conducted on the 8-120B package as required by ANSI N14.5:

Table 4.2
Leakage Tests of the 3-60B Package

<table>
<tr><th>Test</th><th>Frequency</th><th>Test Gas</th><th>Acceptance Criteria</th><th>Procedure</th></tr>
<tr><td>Maintenance</td><td>After maintenance, repair (such as weld repair), or replacement of components of the containment system.</td><td rowspan="3">R-134a, or helium (optional)</td><td rowspan="3">$\leq L_R{}^*$</td><td>8.2.2.1</td></tr>
<tr><td>Fabrication</td><td>Prior to first use of the 8-120B.</td><td>8.1.4</td></tr>
<tr><td>Periodic</td><td>Within 12 months prior to each shipment.</td><td>8.2.2.1</td></tr>
<tr><td>Pre-Shipment</td><td>Before each shipment, after the contents are loaded and the package is closed.**</td><td>nitrogen or air (optional)</td><td>sensitivity $\leq 10^{-3}$ ref-cm^3/sec</td><td>8.2.2.2</td></tr>
</table>

* Adjusted for the individual properties of the test gas; sensitivity is $\leq L_R/2$.

**The pre-shipment leak test is not required for contents that meet the definition of low specific activity material or surface contaminated objects in 10CFR71.4, and also meet the exemption standard for low specific activity material and surface contaminated objects in 10CFR71.14(b)(3)(i).

As shown in Table 4.2, the Maintenance, Fabrication, and Periodic leakage tests may be performed using R-134a, or helium as the tracer gas. The acceptance criterion for these tests is the reference air leakage rate, L_R, which is calculated in Section 4.4. An equivalent maximum permissible leakage rate to L_R has been calculated in Sections 4.5 and 4.6 for R-134a and He, respectively, adjusting for individual properties for the gas plus the test pressure and temperature. The equivalent leakage becomes the acceptance criteria for the particular gas being used to perform the test.

The Fabrication leakage tests are performed on the entire containment boundary including the closure lid, the vent port, the cask inner shell and base plate, and associated weldings. The procedure for performing the leakage tests is described in Section 8.1.4.

The Maintenance and Periodic leakage tests are performed on the closure lid and the vent port. The detailed procedure for performing these tests is given in 8.2.2.1, but generally they will be conducted as follows:

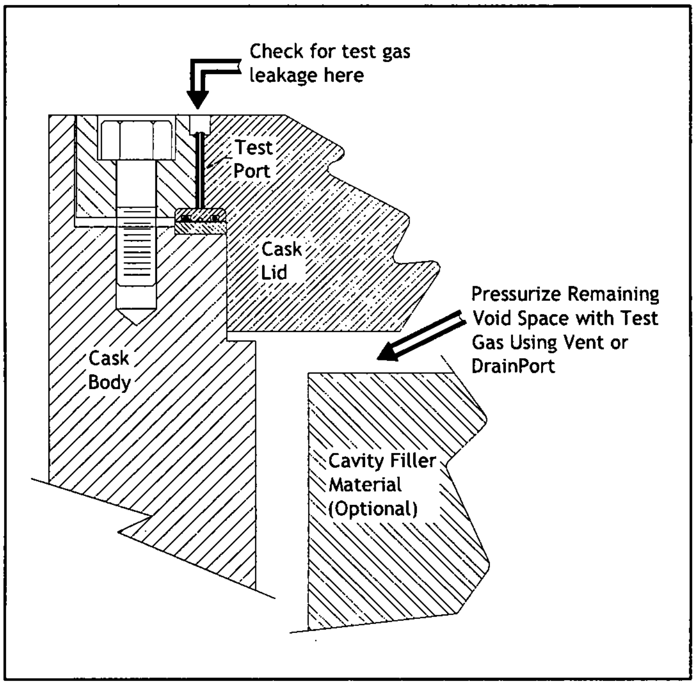


Fig. 4.8 - Periodic Leak Test of Closure Lid

- Pressurize the void space in the cavity with a test gas using the vent port in the lid. Some of the volume of the cavity may be temporarily filled to reduce the volume of test gas required to conduct the test.
- Check for leaks of the inner (containment boundary) O-ring using the test port in the lid.

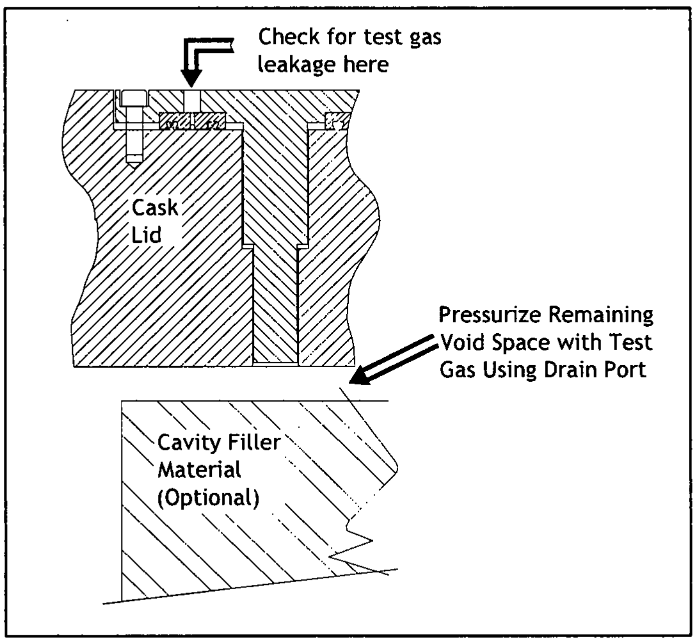


Fig. 4.9 - Periodic Leak Test of Vent Port

- Pressurize the void space in the cavity with a test gas using the drain port. Some of the volume of the cavity may be temporarily filled to reduce the volume of test gas required to conduct the test.
- Check for leaks of the inner (containment boundary) vent port cover plate O-ring

4.9 Periodic Verification Leak Rate Determination for Leaktight Status

4.9.1 Introduction

The purpose of this section is to describe the method for performing a periodic leak test to demonstrate meeting the leaktight criterion per ANSI N14.5-1997. This test method is only applicable to a 8-120B cask with o-rings and seals that meet the helium permeability requirement of Seal Specification ES-C-038 [Ref. 4.4].

4.9.2 Test Conditions

The test is performed with a mass spectrometer leak detector. The test is conducted on the 8-120B by evacuating the cask cavity to at least 90% vacuum then pressurizing the cask cavity with helium (+1 psig, -0 psig). The annulus between the o-rings is evacuated until the vacuum is sufficient to operate the helium mass spectrometer leak detector and the helium concentration in the annulus is monitored. The acceptance criterion is 1.0×10^{-7} atm-cm^3/sec of air (leaktight). The detector sensitivity must be less than or equal to 5.0×10^{-8} atm-cm^3/sec. Similar tests are performed on the vent port.

4.10 References

4.1 American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials, American National Standards Institute, Inc., New York, ANSI N14.5-1997, 1997.

4.2 8-120B Drawing, C-002-12CV01-001, Energy*Solutions*, 2010

4.3 Containment Analysis for Type B Packages Used to Transport Various Contents, LLNL, NUREG/CR-6487, 1996

4.4 ES Specification, ES-C-038, Seal Specification for the 8-120B Cask

Appendix 4.1

Properties of R-134a

Technical Information

P134a

DuPont™ Suva®
refrigerants

DuPont HFC-134a

Properties, Uses, Storage, and Handling

DuPont™ Suva® 134a refrigerant
DuPont™ Suva® 134a (Auto) refrigerant
DuPont™ Formacel® Z-4 foam expansion agent
DuPont™ Dymel® 134a aerosol propellant

DUPONT®

The miracles of science™

Physical Properties of HFC-134a

Physical Properties	Unit	HFC-134a
Chemical Name		Ethane, 1,1,1,2-Tetrafluoro
Chemical Formula		CH_2FCF_3
Molecular Weight	—	102.03
Boiling Point at 1 atm (101.3 kPa or 1.013 bar)	°C °F	–26.1 –14.9
Freezing Point	°C °F	–103.3 –153.9
Critical Temperature	°C °F	101.1 213.9
Critical Pressure	kPa lb/in² abs	4060 588.9
Critical Volume	m³/kg ft³/lb	1.94×10^{-3} 0.031
Critical Density	kg/m³ lb/ft³	515.3 32.17
Density (Liquid) at 25°C (77°F)	kg/m³ lb/ft³	1206 75.28
Density (Saturated Vapor) at Boiling Point	kg/m³ lb/ft³	5.25 0.328
Heat Capacity (Liquid) at 25°C (77°F)	kJ/kg·K or Btu/(lb) (°F)	1.44 0.339
Heat Capacity (Vapor at Constant Pressure) at 25°C (77°F) and 1 atm (101.3 kPa or 1.013 bar)	kJ/kg·K or Btu/(lb) (°F)	0.852 0.204
Vapor Pressure at 25°C (77°F)	kPa bar psia	666.1 6.661 96.61
Heat of Vaporization at Boiling Point	kJ/kg Btu/lb	217.2 93.4
Thermal Conductivity at 25°C (77°F) Liquid	W/m·K Btu/hr·ft°F	0.0824 0.0478
Vapor at 1 atm (101.3 kPa or 1.013 bar)	W/m·K Btu/hr·ft°F	0.0145 0.00836
Viscosity at 25°C (77°F) Liquid	mPa·S (cP)	0.202
Vapor at 1 atm (101.3 kPa or 1.013 bar)	mPa·S (cP)	0.012
Solubility of HFC-134a in Water at 25°C (77°F) and 1 atm (101.3 kPa or 1.013 bar)	wt%	0.15
Solubility of Water in HFC-134a at 25°C (77°F)	wt%	0.11
Flammability Limits in Air at 1 atm (101.3 kPa or 1.013 bar)	vol %	None
Autoignition Temperature	°C °F	770 1,418
Ozone Depletion Potential	—	0
Halocarbon Global Warming Potential (HGWP) (For CFC-11, HGWP = 1)		0.28
Global Warming Potential (GWP) (100 yr ITH. For CO_2, GWP = 1)		1,200
TSCA Inventory Status	—	Reported/Included
Toxicity AEL* (8- and 12-hr TWA)	ppm (v/v)	1,000

* AEL (Acceptable Exposure Limit) is an airborne inhalation exposure limit established by DuPont that specifies time-weighted average concentrations to which nearly all workers may be repeatedly exposed without adverse effects.

Note: kPa is absolute pressure.

SUVA®

REFRIGERANTS

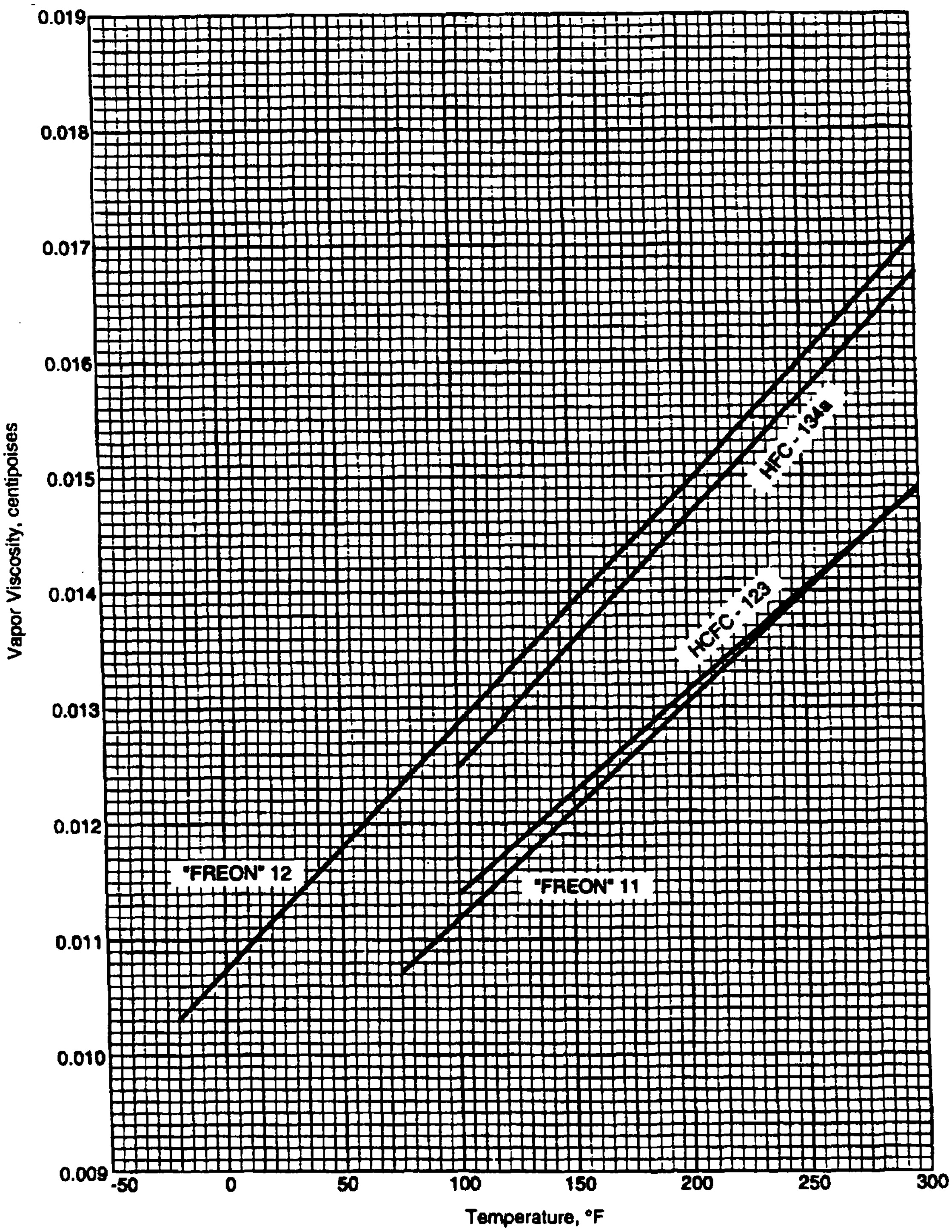
PRODUCT INFORMATION

Transport Properties of SUVA® Refrigerants:

SUVA® 134a
SUVA® 123

Viscosity
Thermal Conductivity
and
Heat Capacity
for the
Liquid and Vapor

Vapor Viscosity at Atmospheric Pressure



5.0 SHIELDING EVALUATION

5.1 Description of Shielding Design

The Model 8-120B packaging consists of a lead and steel containment vessel which provides the necessary shielding for the various radioactive materials to be shipped within the package. (Refer to Section 1.2.3 for packaging contents.) Tests and analysis performed under chapters 2.0 and 3.0 have demonstrated the ability of the containment vessel to maintain its shielding integrity under normal conditions of transport. Prior to each shipment, radiation readings will be taken based on individual loadings to assure compliance with applicable regulations as determined in 10CFR71.47 (see Section 7.1, step 13c).

The 8-120B will be operated under "exclusive use" such that the contents in the cask will not create a dose rate exceeding 200 mrem/hr on the cask surface, or 10 mrem/hr at two meters from the outer lateral surfaces of the vehicle. The package shielding must be sufficient to satisfy the dose rate limit of 10CFR71.51(a) (2) which states that any shielding loss resulting from the hypothetical accident will not increase the external dose rate to more than 1000 mrem/hr at one meter from the external surface of the cask.

5.1.1 Shielding Design Features

The cask side wall consists of an outer 1.5 inch thick steel shell surrounding 3.35 inches of lead and an inner containment shell wall of 0.75 inch thick steel. Total material shield thickness is 2.25 inches of steel and 3.35 inches of lead.

The primary cask lid consists of two layers of 3.25 inch thick steel, giving a total material shield thickness of 6.5 inches of steel. This lid closure is made in a stepped configuration to eliminate radiation streaming at the lid/cask body interface.

A secondary lid is located at the center of the main lid, covering a 29.0 inch opening. The secondary lid is constructed of two 3.25 inch steel plates with multiple steps machined in the secondary lid. These match steps in the primary lid, eliminating radiation streaming pathways.

5.1.2 Maximum Radiation Levels

Table 5.1 gives both normal and accident condition dose rates for the maximum activity Co-60 and Cs-137 source in the cask.

Table 5.1 - Summary of Maximum Dose Rates (mrem/hr)

	Package Surface		1 m from Surface		2m from 8' trailer
Condition	Side	Top/Bottom	Side	Top/Bottom	Side
NCT					
Co-60 Source	181.5	190.0	NA	NA	7.1
Cs-137 Source	119.2	190.0	NA	NA	3.8
Allowable	200	200	NA	NA	10.0
HAC					
Co-60 Source	NA	NA	1000.0	74.8	NA
Cs-137 Source	NA	NA	1000.0	80.6	NA
Allowable	NA	NA	1000.0	1000.0	NA

The following assumptions were used to develop the values given in the table.

5.1.2.1 **Normal Conditions**

The source is modeled as a point source (1 cm dia x 1 cm high) in the top corner of the cavity. Reference 5.7.2 includes a complete summary of the package response functions for all source configurations of interest.

5.1.2.2 **Accident Conditions**

(1) Lead slump of 0.15" resulting from the accident drop analysis is incorporated in the model

(2) Thinning of the lead shield layer due to the puncture drop is incorporated by reducing the lead thickness by 0.5"

(3) The source is modeled as a point source (1 cm dia x 1 cm high) in the top corner of the cavity. Reference 5.7.2 includes a complete summary of the package response functions for all source configurations of interest.

5.1.2.3 **Conclusion**

For the Co-60 corner point source case, the maximum allowable payload gamma source strength is governed by the 200 mrem/hr dose rate limit that applies on the cask body side, under NCT. The results determine a maximum allowable source strength of 3.34 x 10^{11} gammas/sec (5.0 Ci) for that isotope. At this source strength, the results show a dose rate of close to 200 mrem/hr on the package side surface, and dose rates that are well under their regulatory limits at all other locations. Because of a 5% administrative source limit, the dose rate shown in Table 5.1 is 190.0 mrem/hr.

For the Cs-137 corner point source case, the maximum allowable payload gamma source strength is also governed by the 200 mrem/hr dose rate limit that applies on the cask body side, under NCT. The results determine a maximum allowable source strength of 2.36 x 10^{12} gammas/sec (75 Ci) for that isotope. At this source strength, the results show a dose rate of close to 200 mrem/hr on the package surface, and dose rates that are well under their regulatory limits at all other locations. Because of a 5% administrative source limit, the dose rate shown in Table 5.1 is 190.0 mrem/hr.

As the results do not exceed the allowable dose rates, the 8-120B cask meets the shielding requirements of 10 CFR Part 71.

5.2 Source Specification

5.2.1 Gamma Source

Analyses are performed for idealized source configurations that bound any actual source configuration that may occur. These bounding configurations are: a point source at the center of the cask cavity in the NCT configuration, a point source at the top corner of the cask cavity in the NCT configuration, a point source in the top corner of the cask cavity in the HAC configuration, and a uniform mass of material within a defined source region, as described in Section 5.4, for both NCT and HAC configurations. Further details of the analyses are found in Ref. 5.7.2.

All of the analyses described above are performed for several gamma energy levels, ranging from 0.5 MeV to 3.5 MeV. Two specific isotope cases, Co-60 and Cs-137 (and the corresponding specific gamma energies) are also analyzed. The Cs-137 source includes an equilibrium amount of Ba-137m. The photon energy and abundance of Co-60 and Cs-137 are shown in Table 5.2.

Table 5.2 – Photon Energy and Abundance

Radionuclide	**Photon Energy**	**Abundance**
	MeV	**# of Gamma/decay**
^{60}Co	1.176	1
	1.333	1
^{137}Cs	0.662	0.85

5.2.2 Neutron Source

There are no significant sources of neutron radiation in the radioactive materials carried in the CNS 8-120B cask that result in measureable neutron doses outside the cask.

5.2.3 Beta Source

Significant beta emitters may be qualified as equivalent gammas as described in Section 5.4.4.

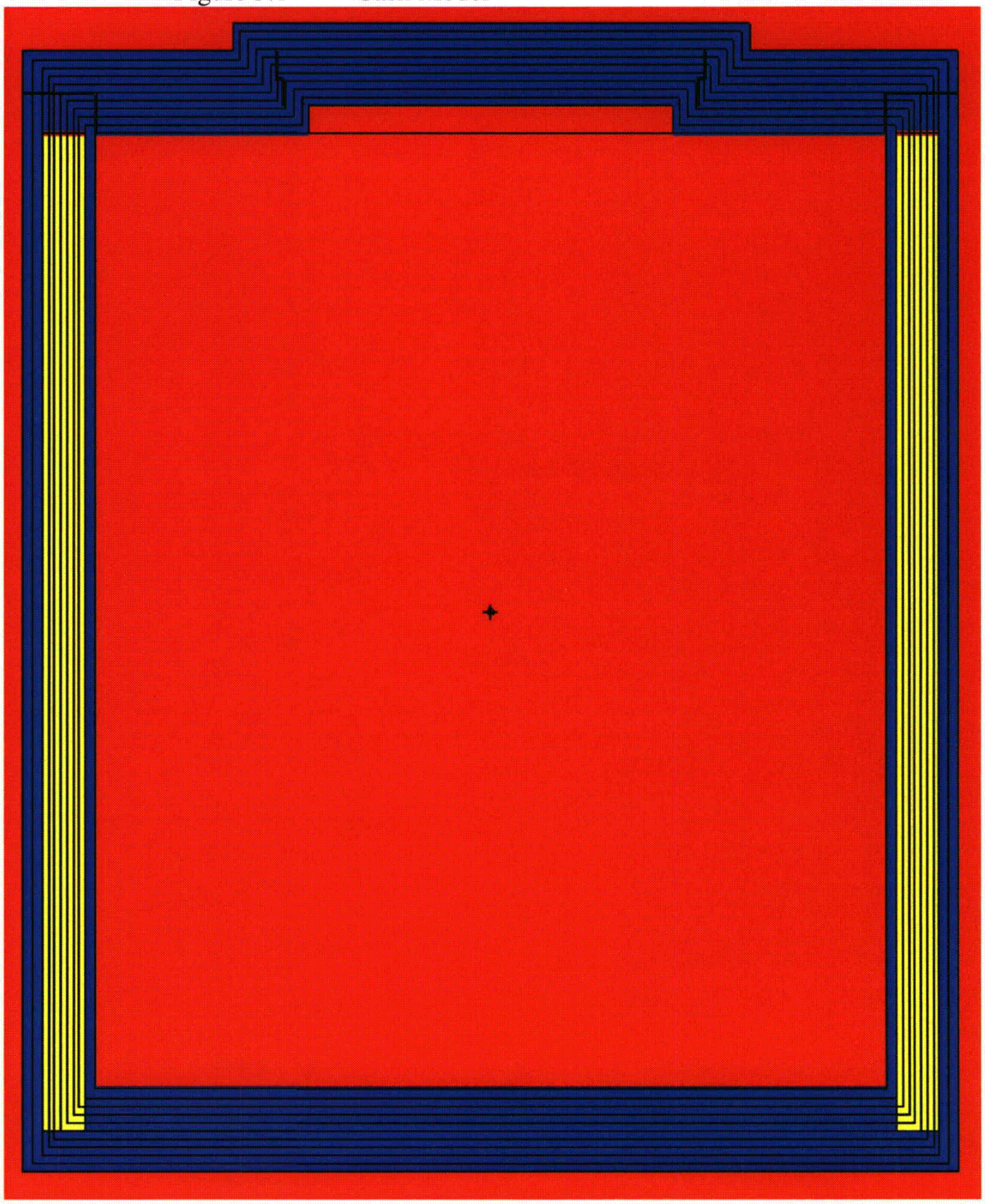
5.3 Model Specification

5.3.1 Description of Radial and Axial Shielding Configuration

Normal Conditions of Transport (NCT)

The walls of the 8-120B cask, 0.75" inner and 1.5" outer steel walls, with a 3.35" lead layer between, are modeled as cylindrical shells around the cavity cylinder. The base and lid of the cask are two 3.25" steel plates, for a total thickness of 6.5". This geometry is shown in Figure 5.1; the impact limiters are not shown. The cask is transported upright, i.e., with the axis of the cylinder vertical. Doses are evaluated at contact with the cask sidewall, the impact limiter surface, and at 2m from the 8' wide trailer.

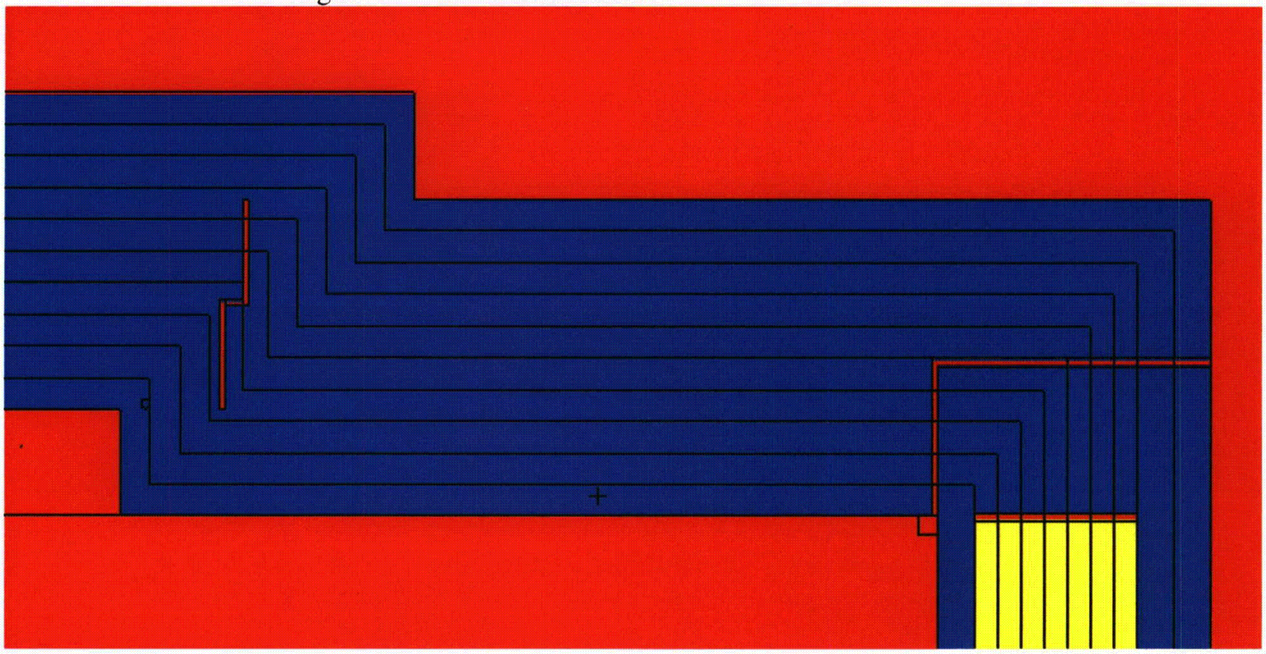
Figure 5.1 Cask Model



Hypothetical Accident Conditions (HAC)

As discussed in Chapter 2, the hypothetical accident 30' drop results in a 0.15" lead slump and the puncture drop causes a local ½" thinning of the lead layer. The HAC model has a 0.15" air-filled void at the top of the lead shield layer. Also, to conservatively reflect the puncture drop thinning, the thickness of the radial lead shield is reduced by ½" in the HAC model. The impact limiters are conservatively ignored. The HAC model is shown in Figure 5.2. Doses are determined at 1 m from the sidewall and the lid.

Figure 5.2 – HAC Cask Model



See Reference 5.7.2 for additional details of the MCNP models.

5.3.2 **Material Properties**

The compositions and densities of the materials modeled in the shielding analyses are described in Table 5.3 below. The table also lists the MCNP material/cross-section identifier (ZAID) for each modeled material. The density of air is taken from Input #3.are shown in below.

Table 5.3 – Material Composition and Density

Material	Total Density (g/cc)	Composition	MCNP ZAID
Carbon Steel	7.82	99% Fe 1% C	26000.84p 8000.84p
Lead	11.34	100% Pb	82000.84p
Air	0.001205	76.508% N 23.479% O 0.013% C	7000.84p 8000.84p 6000.84p

5.4 **Shielding Evaluation**

The 8-120B package carries a range of contents, from small concentrated sources to large volume homogeneous materials and combinations of these, and may include nearly every radionuclide. In order to determine the maximum activity of any particular radionuclide or mixture of radionuclides, a series of evaluations of bounding source configurations over a range of gamma energies are performed to determine the maximum activity (γ/sec) or maximum activity density (γ/sec·g) for each combination of configuration and energy that results in the meeting the most restrictive of the dose rate limits from 10 CFR 71.47 and §71.51. The resulting set of activity limits ensure that any content meeting the activity limit for the appropriate configuration and gamma energy will comply with the §71.47 and §71.51 limits.

5.4.1 Methods

The gamma dose rates were calculated using MCNP Version 5, rev. 1.51.

In addition to the point source locations noted in Section 5.2, a uniformly-distributed gamma source is modeled within the source region. The uniform mass that fills the defined source region is zirconium or aluminum, whichever has the more conservative (smaller) attenuation coefficient at the gamma energy thus bounding other contents materials. The uniform mass is set at a density of 9.0 g/cc, which exceeds the density of nearly all expected payloads. Since the distributred source analyses determine limits in activity density (γ/sec-gram), this density bounds all other lower density contents. Defined source regions include the entire cask interior cavity, a "55 gallon" source zone centered within the cavity and a 2.5 ft^3 source zone centered within the cavity. All the above source zones are modeled for the NCT cask configuration. For the HAC cask configuration, only the full-cask-cavity source zone is modeled.

For the normal condition of transport (NCT) cases, dose rates are tallied on the vertical surface two meters from the package/transporter side (i.e., 322 cm from the cask centerline), and on the package surface which includes the impact limiter side and end surfaces as well as the cask body side cylindrical surface that lies between the impact limiters.

For the HAC point source cases, the dose rates are tallied at two locations on the surface one meter from the cask body. One location lies on the radial one meter surface, directly across from the source point (viewing the source point through the lead slump gap). The second location lies on the top one meter surface, directly above the source point, viewing the source point through the gap between the radial cask body and the lower part of the primary cask lid.

For the HAC distributed source cases, the dose rates are tallied over the entire spans of the surfaces that lie one meter from the side, top and bottom of the cask body.

For each of the analyses, the peak dose rates (per source gamma) that occur on each of the (NCT or HAC) regulatory surfaces described above are determined.

From these peak dose rates, limits are calculated over the range of gamma energies 0.5-3.5 MeV and for the radionuclides Co-60 and Cs-137. The limits are determined, in activity (γ/sec) for the point source configurations and in activity density (γ/sec·g) for the distributed source cases. The regulatory dose rate limit for each surface is divided by the highest per-source-gamma dose rate for that surface, to yield a maximum source strength, in γ/sec or γ/sec·g. The lowest of the allowable source strengths is then selected as the limiting gamma source strength for that case. Then, for the distributed source cases (only), the allowable source strength is divided by the modeled source region mass to yield the allowable source strength density in γ/sec·g.

5.4.2 Input and Output Data

The MCNP input and output files are found in Reference 5.7.3. The input file lists the inputs that define the source dimensions, shield dimensions, materials and density, and source spectrum.

5.4.3 Flux-to-Dose-Rate Conversion

The flux to exposure rate conversion factors are listed in Table 5.4 (Ref. 5.7.1).

Table 5.4 - Gamma-Ray-Flux-To-Dose-Rate Conversion Factors (ANSI/ANS-1977)

Gamma Energy (MeV)	DCV (rem/hr) per (γ/cm^2-sec)
0.015	1.95E-06
0.025	8.01E-07
0.045	3.17E-07
0.08	2.61E-07
0.15	3.79E-07
0.30	7.59E-07
0.50	1.15E-06
0.65	1.44E-06
0.75	1.60E-06
0.90	1.83E-06
1.25	2.32E-06
1.75	2.93E-06
2.5	3.72E-06
3.5	4.63E-06
4.5	5.42E-06
5.5	6.19E-06
6.5	6.93E-06
7.5	7.66E-06
9.0	8.77E-06
12.0	1.10E-05

5.4.4 External Radiation Levels and Activity Limits

Gamma Activity Limits

The results of the analyses of the bounding configurations are compared to the external radiation limits allowed for the various compliance locations identified in §71.47 and §71.51. The configuration, at each energy, that has the largest ratio of result to limit is set as the governing configuration from which the limits are established.

The final results of the shielding evaluation are the limits on payload gamma source strength (γ/sec) and payload gamma source strength density (γ/sec·g), which vary as a function of gamma energy and payload configuration. These limits are presented, for all gamma energies and all analyzed source configurations, in Table 5.5 below. The limits are presented graphically in Figure 5.3 and Figure 5.4.

Table 5.5 - Final Payload Source Strength and Source Strength Density Limits

Energy (MeV)	General Sources		Discrete Sources (shored at centroid)*		
	Source γ/sec	Source Density γ/sec·g	Source γ/sec	Source Density γ/sec·g	
				2.5 ft^3	55 gal
	❶	❷	❸	❹	❺
3.50	3.276E+10	3.529E+05	2.234E+11	2.385E+06	1.265E+06
2.75	4.746E+10	5.516E+05	3.009E+11	3.710E+06	1.970E+06
2.25	7.088E+10	8.933E+05	4.332E+11	5.806E+06	3.095E+06
1.83	1.143E+11	1.709E+06	7.317E+11	1.074E+07	5.761E+06
1.50	2.012E+11	4.030E+06	1.155E+12	2.416E+07	1.304E+07
1.17	4.023E+11	1.104E+07	1.933E+12	7.974E+07	5.255E+07
0.90	8.725E+11	3.475E+07	3.539E+12	2.563E+08	1.586E+08
0.70	1.859E+12	1.199E+08	6.238E+12	8.658E+08	5.086E+08
0.50	6.369E+12	8.392E+08	1.481E+13	5.893E+09	3.286E+09
Co-60	3.336E+11	8.082E+06	1.652E+12	5.789E+07	3.312E+07
Cs-137	2.362E+12	1.607E+08	7.150E+12	1.173E+09	6.674E+08

**For discrete source limits, use columns ❸ and ❹ when the payload object meets the 2.5 ft^3 size criteria, or columns ❸ and ❺ when it meets the 55 gallon size criteria. When the size meets neither criteria use columns ❶ and ❷.*

The "general" source limits shown in the left side of Table 5.5 apply for payloads that fill most of the cask cavity or are not shored within a smaller volume at the cavity center. The discrete source limits shown in the right part of Table 5.5 may apply if the payload meets the size criteria and is shored to the center of the cask cavity.

Detail of the calculations (and process) used to determine the payload source limits shown in Table 5.5 are found in Ref. 5.7.2.

Figure 5.3 - Payload Gamma Source Strength Limit vs. Gamma Energy

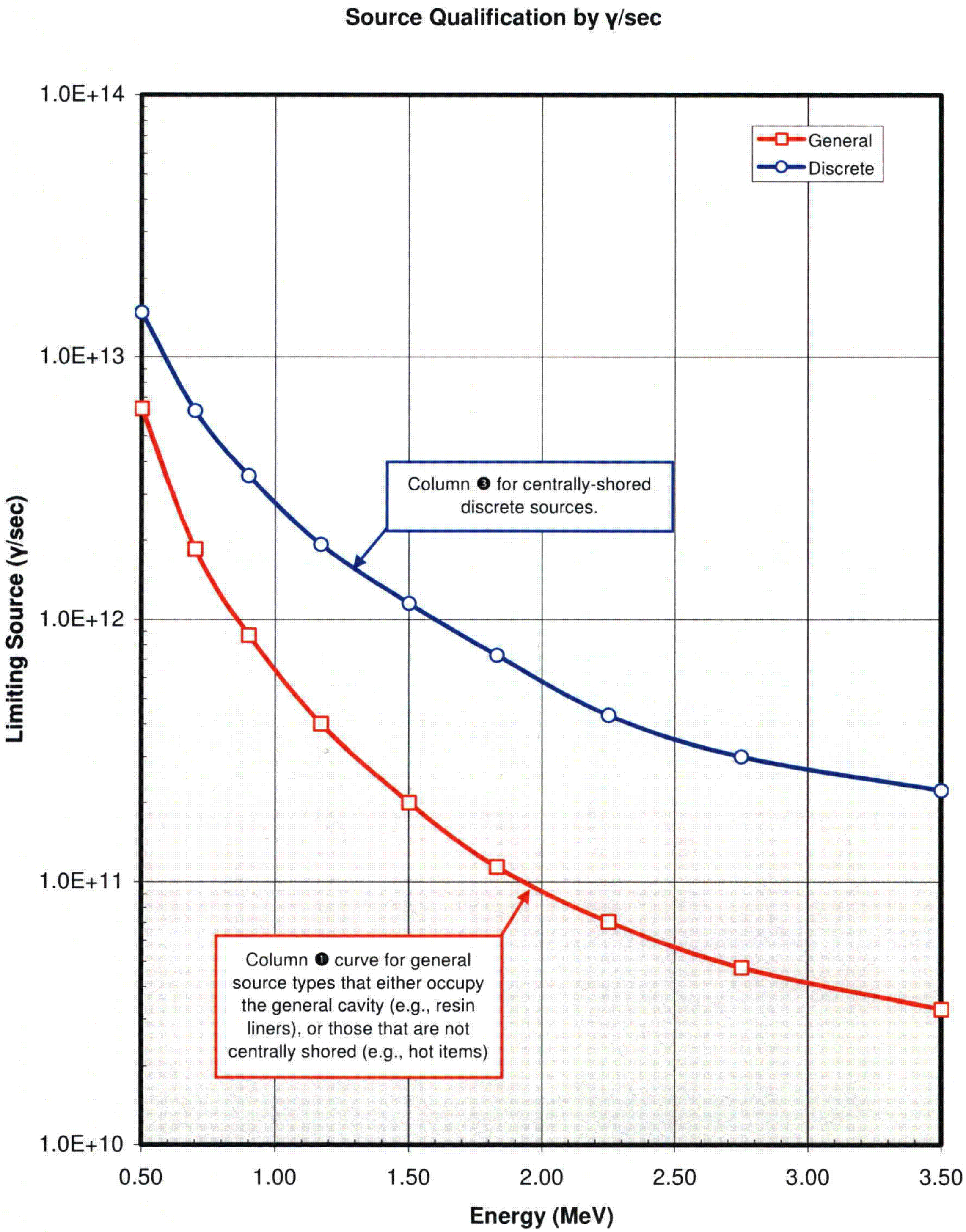
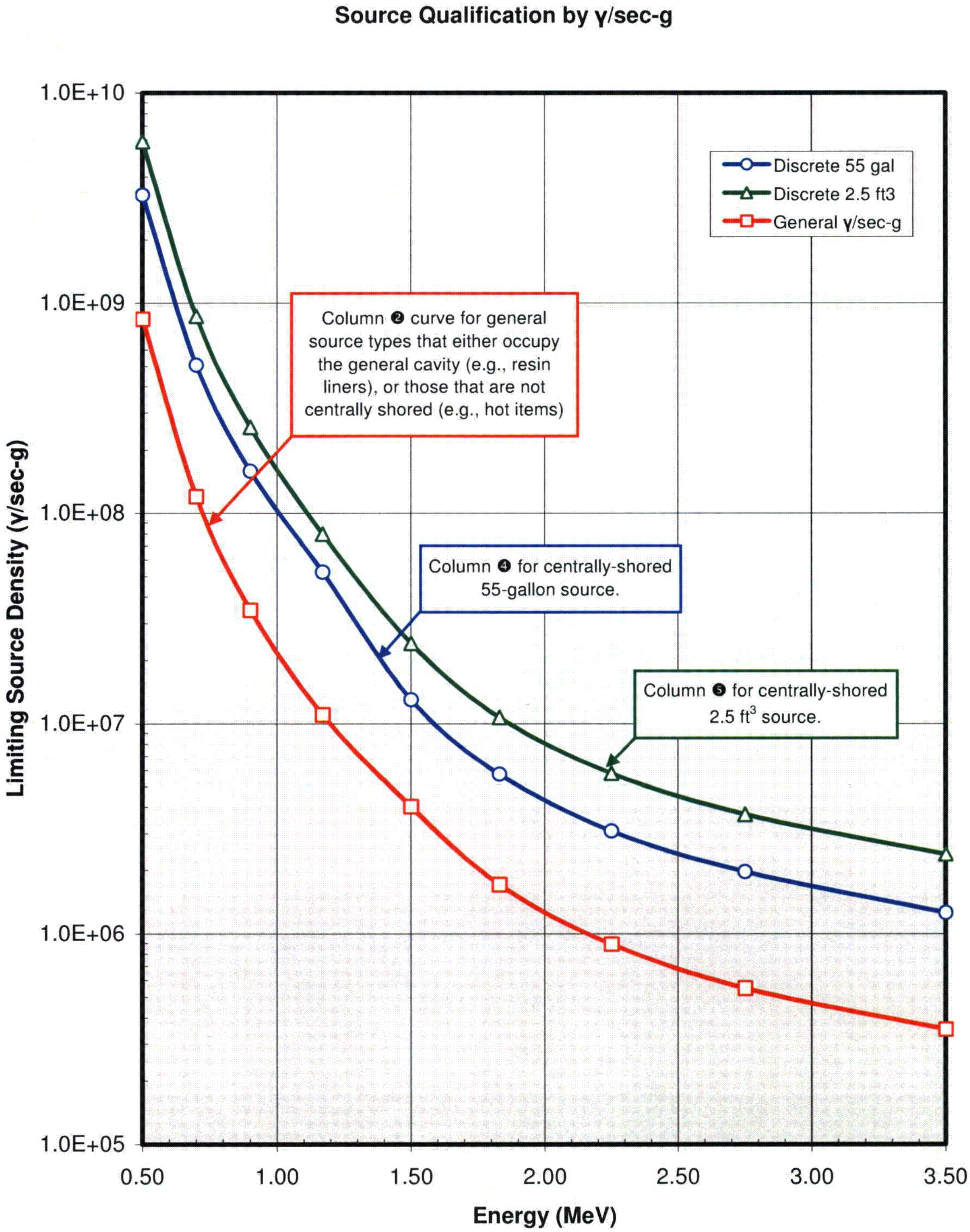


Figure 5.4 - Payload Gamma Source Strength Density Limit vs. Gamma Energy



Beta Activity Limits

Beta particles lose their energy continuously as they pass through matter, emitting Bremsstrahlung photons over their range. These Bremsstrahlung photons, however, have the potential to be significant contributors to package dose rates because the allowable (3000 A_2) source activity for betas can be much higher than for gamma emitters (e.g., as much as 42,000 Ci of 32P vs. 4144 Ci of 137Cs). The method for qualifying significant 8-120B beta emitters is to represent the beta emitter as an equivalent photon emitter and treat it like any other photon energy line per the methods described in 5.5. Since the equivalent photon activity is less than 1% of the beta activity, this method is only applied to beta sources with activities greater than 10 times the most restrictive gamma source limit or 2E+12 betas per second. See Ref. 5.7.2 for additional details and validating calculations.

The beta source can be converted to an equivalent photon source by:

$$S_\lambda = S_\beta \cdot \frac{S_\lambda}{S_\beta} \text{ where}$$

S_γ = equivalent monoenergetic gamma source, γ/sec, at the maximum beta energy E_{max}.
S_β = beta source, β/sec, at the beta energy spectrum for the nuclide of interest

and

$$\frac{S_\lambda}{S_\beta} = (\textit{fraction of energy converted from betas to photons})\left(\frac{\textit{beta } E_{avg}}{\textit{photon energy}}\right)$$

Conservatively assume all photons are at the beta maximum energy E_{max}, the energy ratio becomes:

$$\frac{S_\lambda}{S_\beta} = f\left(\frac{E_{avg}}{E_{max}}\right)$$

where

E_{avg}	=	average energy of the beta source distribution, MeV
E_{max}	=	maximum energy of the source distribution, MeV.

The fraction of the incident beta energy that is converted to photon energy, f, is given by (Ref. 5.7.3)

$$f \cong 3.5 \times 10^{-4} ZE_{max}$$

where

f	=	the fraction of the incident beta energy that is converted to photon energy,
Z	=	atomic number of the absorber

So

$$\frac{S_\lambda}{S_\beta} = 3.5\times10^{-4} ZE_{max}\left(\frac{E_{avg}}{E_{max}}\right)$$

The resulting equation to convert a beta source to an equivalent photon source at the beta's maximum energy is therefore:

$$S_\lambda = S_\beta\left(3.5\times10^{-4} ZE_{avg}\right)$$

For a single material absorber, use the Z of the material (effective Z for a compound). For mixtures, use a weighted average Z_w:

$$Z_W = \sum_{i=1}^{n}\left(\frac{m_i}{m_{total}} \cdot Z_i\right)$$

The proposed method for qualifying significant 8-120B beta emitters is to represent the beta emitter as an equivalent photon emitter and treat it like any other photon energy line per the methods described in the remainder of this calculation. In this way, significant beta emitters can be accounted for along with other gamma emitters.

A procedure for evaluating beta emitters is included in Chapter 7 Attachment 1 which establishes limits for large activity beta sources.

5.5 Payload Qualification

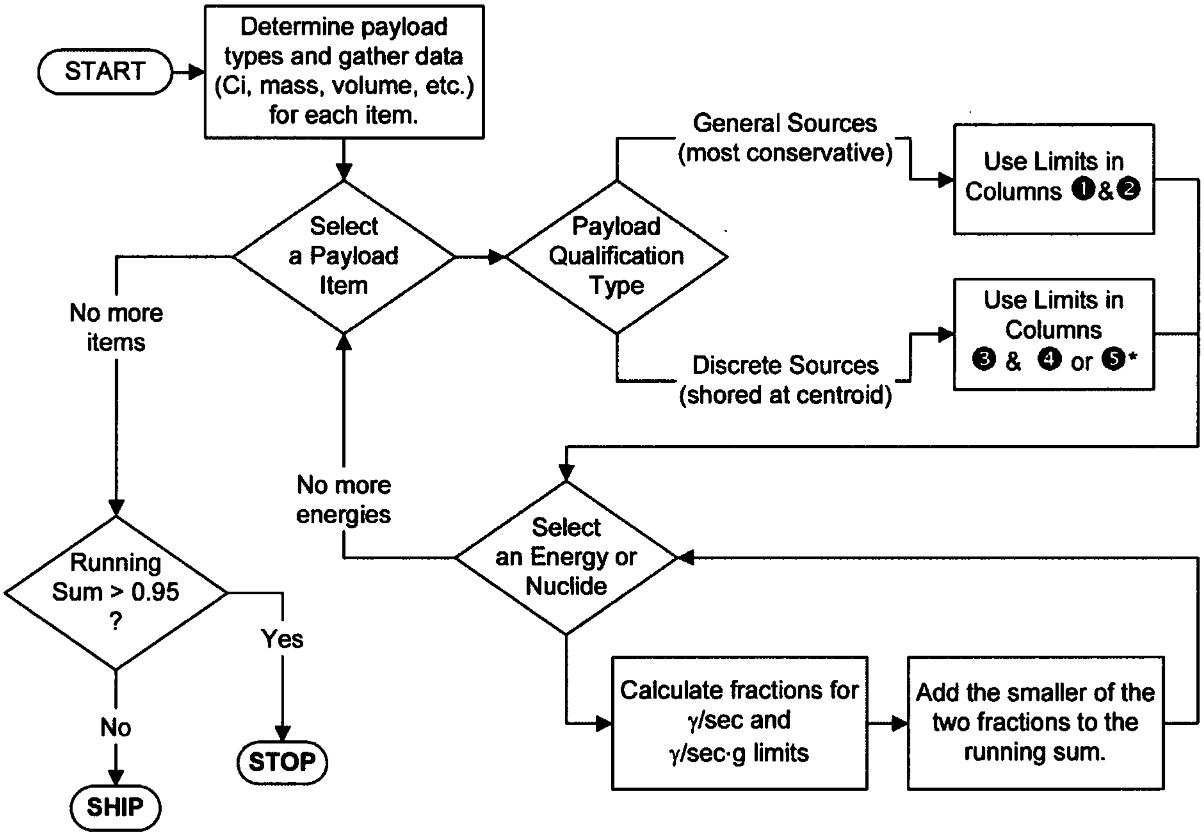
Radioactive 8-120B contents must be qualified to ensure the shipment will meet the regulatory dose limits from §71.47 and §71.51

To qualify a payload, the cask user determines 1) a gamma source strength (γ/sec) and 2) a gamma source strength density (γ/sec·g) for their payload, based on the gamma energy that applies for the payload, whether the payload is shored at the cavity centroid, and the size and volume of the payload. The payload qualifies for shipment in the 8-120B cask if it meets either one of the source strength or source strength density limits in Table 5.5. Note that when determining compliance with the source strength density limit, the highest source strength density (or "hottest") section of the waste must be used.

To qualify payloads that emit gammas at multiple energies or when portions of the payloads are radiologically different, a sum of fractions approach is used. For multiple payload types, the user performs a separate qualification evaluation for each payload component/energy, and then use a sum of fractions approach to qualify the overall cask payload. For each gamma energy or payload component, two fractions are determined, one based on the ratio of the payload source strength (γ/sec) over the allowable source strength, and one based on the ratio of the source strength density (γ/sec·g) over the allowable source strength density. The lower of the two fractions is then selected, for each gamma energy or payload component. The resulting fractions are then summed. The total (sum of fractions) may not exceed 0.95.

This qualification process is shown in the flowchart below (Figure 5.5)

Figure 5.5 - Payload Gamma Source Strength Density Limit vs. Gamma Energy



5.6 Conclusion

The cask shielding must be able to limit the dose rate to the limits of §71.47 and §71.51. This section demonstrates compliance with this requirement. Structural analysis (Section 2.0) demonstrates that the cask wall will not fail during the hypothetical accident. However, lead slump may occur during a drop giving an isolated region in the sidewall without lead. Lead slump cannot occur in the lid or bottom of the cask since lead is not present in these parts of the cask. With application of the activity qualification process from Section 5.5, the contents will meet the dose rate limits.

5.7 References

5.7.1 ANSI/ANS 6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors."
5.7.2 ES Calculation, NU-391 Rev. 2, "8-120B Shielding Response"
5.7.3 Cember, H., "Introduction to Health Physics," Pergamon Press, 2nd Ed.

7.0 OPERATING PROCEDURE

This chapter describes the general procedure for loading and unloading of the 8-120B Cask.

The maximum permissible activity is the lesser of the activity determined by:
1) Attachment 1 for beta and gamma emitters, 2) 3000 A_2, or 3) having a decay heat of 200 watts. Radioactive contents are to be transported as exclusive use, per 10 CFR 71.4.

For contents that could radiolytically generate combustible hydrogen, see Attachment 2 for instructions on determination of hydrogen concentration.

Powdered solids shipments require the cask to be leaktight. The most recent periodic leak test must meet the requirements of Chapter 4, Section 4.9, Periodic Verification Leak Rate Determination for Leaktight Status.

7.1 Loading the Packaging

Note: Prior to loosening the impact limiter ratchet binders, inspect the exterior of the cask for damage, e.g., large dents, gouges, tears to the impact limiter skin and thermal shield. Contact EnergySolutions if damage is present. The cask may not be used as a Type B package until the damage is assessed by Energy*Solutions* and repairs, if required, are made to achieve conformance with the licensing drawings.

7.1.1 Loosen and disconnect ratchet binders from upper impact limiter.

7.1.2 Using suitable lifting equipment, remove upper impact limiter assembly. Care should be exercised to prevent damage to impact limiter during handling and storage.

7.1.3 Determine if cask must be removed from trailer for loading purposes. To remove cask from trailer:

7.1.3.1 Disconnect cask to trailer tie-down equipment.

7.1.3.1.1 Inspect cask lifting ear bolts for defects. Obtain replacement bolts as specified on the drawing listed in 5(a)(3) of the CoC for any bolts that show cracking or other visual signs of distress.

7.1.3.1.2 Inspect cask lifting ear threaded holes for defects. Contact Energy*Solutions* if any bolt holes show signs of cracking or visual signs of distress.

7.1.3.2 Attach cask lifting ears and torque bolts to 200 ft-lbs. ± 20 ft-lbs. lubricated.

NOTE: The cables used for lifting the cask must have a true angle, with respect to the horizontal of not less than 60°.

7.1.3.3 Using suitable lifting equipment, remove cask from trailer and the lower impact limiter and place cask in level loading position.

NOTE: In certain circumstances, loading may be accomplished through the secondary lid, into a pre-positioned waste liner that has been properly shored or into pre-positioned shoring, while the primary lid remains on the cask. Alternate "(A)" steps have been included to accommodate this situation.

7.1.4 Loosen and remove the twenty (20) bolts, which secure the primary lid to cask body.

7.1.4A Loosen and remove the twelve (12) bolts, which secure the secondary lid to the primary lid.

7.1.5 Inspect the bolts for defects. Obtain replacement bolts as specified on the drawing listed in 5(a)(3) of the CoC for any bolts that show cracking or other visual signs of distress.

NOTE: The cables used for lifting either lid must have a true angle, with respect to the horizontal, of not less than 45°.

7.1.6 Remove primary lid from cask body using suitable lifting equipment. Care should be taken during lid handling operations to prevent damage to cask or lid seal surfaces.

7.1.6A Remove secondary lid from cask body using suitable lifting equipment. Care should be taken during lid handling operations to prevent damage to cask or lid seal surfaces.

7.1.7 Inspect the bolts holes for defects. Contact Energy*Solutions* for any bolt holes that show signs of cracking or visual signs of distress.

7.1.8 Inspect cask interior for damage, loose materials or moisture. Clean and inspect seal surfaces. Replace seals when defects or damage is noted which may preclude proper sealing. Contact EnergySolutions if damage is present.

NOTE: Radioactively contaminated liquids may be pumped out or removed by use of an absorbent material. Removal of any material from inside the cask shall be performed under the supervision of qualified health physics personnel with the necessary H.P. monitoring and radiological health safety precautions and safeguards.

NOTE: When seals are replaced, leak testing is required as specified in section 8.2.2.1.

NOTE: Verify intended contents meet the requirements of the Certificate of Compliance.

NOTE: Ensure the contents, secondary container, and packaging are chemically compatible, i.e., will not react to produce flammable gases.

7.1.9 Place disposable liner, drums or other containers into the pre-positioned shoring and install additional shoring or bracing, if necessary, to restrict movement of contents during normal transport.

7.1.9A Process liner as necessary, and cap using standard capping devices.

7.1.10 Clean and inspect lid seal surfaces.

7.1.11 Replace the primary lid on the cask body. Secure the lid by hand tightening the twenty (20) primary lid bolts.

7.1.11.1 Torque, using a star pattern, the twenty (20) primary lid bolts (lubricated) to 250 ft-lbs. ± 25 ft-lbs.

7.1.11.2 Re-Torque, using a star pattern, the twenty (20) primary lid bolts (lubricated) to 500 ft-lbs. ± 50 ft-lbs.

7.1.11A Replace the secondary lid on the primary lid. Secure the lid by hand tightening the twelve (12) secondary lid bolts.

7.1.11.1A Torque, using a star pattern, the twelve (12) secondary lid bolts (lubricated) to 250 ft-lbs. ± 25 ft-lbs.

7.1.11.2A Re-torque, using a star pattern, the twelve (12) secondary lid bolts (lubricated) to 500 ft-lbs. ± 5 0 ft-lbs.

7.1.12 Replace the vent port cap screw and seal (if removed) and torque to 20 ft-lbs. ± 2 ft-lbs.

NOTE: Leak test the primary lid and secondary lid O-rings and the vent port in accordance with Section 8.2.2.2, prior to shipment of the package loaded with greater than "Type A" quantities of radioactive material. For content exemptions of this test, refer to the current Certificate of Compliance No. 9168.

7.1.13 If cask has been removed from trailer, proceed as follows to return cask to trailer:

7.1.13.1 Using suitable lifting equipment, lift and position, cask into lower impact limiter on trailer in the same orientation as removed.

7.1.13.2 Unbolt and remove cask lifting ears.

7.1.13.3 Reconnect cask to trailer using tie-down equipment.

7.1.14 Using suitable lifting equipment, lift, inspect for damage, and install upper impact limiter assembly on cask in the same orientation as removed.

7.1.15 Attach and hand tighten ratchet binders between upper and lower impact limiter assemblies.

7.1.16 Cover lift lugs as required.

7.1.17 Inspect package for proper placards and labeling.

7.1.18 Complete required shipping documentation.

7.1.19 Prior to shipment of a loaded package, the following shall be confirmed:

7.1.19.1 That the licensee who expects to receive the package containing materials in excess of Type A quantities specified in 10 CFR 20.1906(a) meets and follows the requirements of 10 CFR 20.1906, as applicable.

7.1.19.2 That trailer placarding and cask labeling meet DOT specifications (49 CFR 172).

7.1.19.3 That the provisions of 10 CFR 71.87 are met including that the external radiation dose rates are less than or equal to 200 millirem per hour (mrem/hr) at the surface and less than or equal to 10 mrem/hr at 2 meters in accordance with 10 CFR 71.47 by performing radiation surveys. These surveys should be sufficient to ensure that a non-uniform distribution of radioactivity does not cause the surface or 2m limit to be exceeded.
The SAR thermal analysis demonstrates that by meeting the

200w decay heat limit, the temperature requirement of 10 CFR 71.43(g) is met. No temperature survey is required.

7.1.19.4 That all security seals are properly installed.

7.1.19.5 Prior to shipping a loaded package, inspect the exterior of the cask for damage, e.g., large dents, gouges, tears to the impact limiter skin and thermal shield. Contact Energy*Solutions* if damage is present.

7.2 Unloading the Package

In addition to the following sequence of events for unloading a package, packages containing quantities of radioactive material in excess of Type A quantities specified in 10 CFR 20.1906(a) shall be received, monitored, and handled by the licensee receiving the package in accordance with the requirements of 10 CFR 20.1906, as applicable. Identification of packages containing greater than Type A quantities can be made by review of the shipping papers accompanying the shipment.

7.2.1 Move the unopened package to an appropriate level unloading area.

7.2.2 Perform an external examination of the unopened package. Record any significant observations.

7.2.3 Remove security seal(s), as required.

7.2.4 Loosen and disconnect ratchet binders from the upper impact limiter assembly.

7.2.5 Remove upper impact limiter assembly using caution not to damage the cask or impact limiter assembly.

7.2.6 If cask must be removed from trailer, refer to Step 7.1.3.

7.2.7 Loosen and remove the twenty (20) primary lid bolts.

<u>NOTE</u>: The cables used for lifting the lid must have a true angle with respect to the horizontal of not less than 45 degrees.

7.2.8 Using suitable lifting equipment, lift lid from cask using care during handling operations to prevent damage to cask and lid seal surfaces.

7.2.9 Remove contents.

NOTE: Radioactively contaminated liquids may be pumped out or removed by use of an absorbent material. Removal of any material from inside the cask shall be performed under the supervision of qualified health physics personnel with the necessary H.P. monitoring and radiological health safety precautions and safeguards.

7.2.10 Assemble packaging in accordance with loading procedure (7.1.10 through 7.1.19).

7.3 Preparation of Empty Packaging for Transport

7.3.1 Confirm the cavity is empty of contents are far as practicable

7.3.2 Survey the interior; decontaminate the interior if the limits of 49 CFR 173.428(d) are exceeded

7.3.3 Install the lid.

7.3.3.1 Install the lid closure bolts.

7.3.3.2 Torque, using a star pattern, the twenty (20) primary lid bolts (lubricated) to 250 ft-lbs. ± 25 ft-lbs.

7.3.3.3 Re-Torque, using a star pattern, the twenty (20) primary lid bolts (lubricated) to 500 ft-lbs. ± 50 ft-lbs.

7.3.4 Re-install the vent port cap screw with the seal. Torque the vent port cap screw to 20±2 ft-lbs.

7.3.5 Decontaminate the exterior surfaces of the cask as necessary.

7.3.6 Inspect the exterior and confirm it is unimpaired.

7.3.7 Install the impact limiters.

7.3.8 Attach the tamper-indicating seals.

7.3.9 Confirm the requirements of 49 CFR 173.428 are met.

Attachment 1
Determination of Acceptable Beta and Gamma Activity
(see Chapter 5 for the derivation of the beta and gamma activity limits)

Background and Definitions

8-120B contents (payloads) have acceptable beta and gamma sources when they can be shown to meet the requirements in Table 1 using the procedure described in this Attachment. Source qualification is based on a sum-of-fractions method, where sources are broken down into separate gamma energy lines and compared to the corresponding limit for that group. For some payloads, it may be necessary to subdivide the payload into separate items, determining fractions for each item by energy group then summing the fractions to determine acceptability.

Table 1 categorizes the limits into source (γ/sec) and source density (γ/sec·g). For each energy, the fraction to be summed is the lowest of the γ/sec and γ/sec·g fractions. Table 1 has five columns of limits, denoted ❶ through ❺. Depending on the nature of the payload, the user must select a pair of columns to use for each payload item, one γ/sec column and one γ/sec·g column. The "general" payload columns (❶,❷) are the most conservative and are suitable for any payload type. Higher limits are acceptable for special cases where the item is shored about the centroid of the package cavity (e.g., an isotope source). These are termed "discrete" payload items, and are distinguished as follows:

- Use the 2.5 ft^3 limits (❸,❹) when the payload item has a volume of 2.5 ft^3 (70,792 cm) or less, a height of 28 inches (71.16 cm) or less, and a diameter of 17.65 inches (44.84 cm) or less, and is shored at the centroid of the cavity.

- Use the 55-gallon limits (❸,❺) when the payload item has a volume of 7.7 ft^3 (218,868 cm^3) or less, a height of 33.5 inches (85.1 cm) or less, and a diameter of 25.7 inches (65.3 cm) or less, and is shored at the centroid of the cavity.

- If the payload item does not meet the requirements of either the 2.5 ft^3 or 55-gallon definitions, regardless of shoring, then use the γ/sec limit for general sources ❶, and the general γ/sec·g limit ❷.

When determining source strengths for the purpose of qualification, the activities shall be based on the total γ/sec for the payload and the γ/sec·g of the highest-activity ("hottest") portion surveyed. This conservative approach ensures that package dose rate limits will be met. For some payloads, use of the highest activity density may be inappropriately conservative (e.g., radiologically non-homogeneous items). The qualification methodology takes these payloads into consideration, and allows for the qualification to be performed "item by item," where items may be any radiologically distinct objects (e.g., resins, activated metal, crud, beta emitters, etc.). The homogeneity of the payload will determine the way in which it must be qualified. Averaging radiologically distinct sources is not allowed, however it is acceptable to treat the payload as if it were made up of entirely the most significant of the sources. Radiologically non-homogenous materials (e.g., contaminated soil with hot "chunks", activated metal with varying activity levels, etc.) may be qualified separately: e.g., for contaminated soil with hot chunks, the components are the soil, and the hotter particles.

Gamma sources below 0.3 MeV may be neglected. Any sources with gamma energies above 3.5 MeV are not qualified at this time. Table 1 has two special rows for the common radioactive nuclides, ^{60}Co and ^{137}Cs; and so their fractions may be calculated directly without breaking them down into their separate energy lines.

Beta emitters are qualified by converting the beta source strength into an equivalent bremsstrahlung (gamma) source and entering that source like any other source line in the sum-of-fractions. Beta sources with maximum beta energies below 0.3 MeV or source strengths less than 2E+12 β/sec may be neglected. The method for converting betas is presented in the procedure below and the methodology is discussed in Chapter 5 of the SAR.

Payload contents with densities between 0.0 and 9.0 g/cc are within the range of validity for Table 1 γ/sec·g limits. Most materials fall within this range, with the exception of lead and some exotic metals. Do not consider liner, or other secondary container, materials when calculating density. Densities are for the basic material, and should not include voids. Radioactive payload items with densities above 9.0 g/cc must be qualified using the γ/sec limits alone.

Table 1 - Payload Source and Source Density Limits

Energy (MeV)	General Sources		Discrete Sources (shored at centroid)		
	Source γ/sec	Source Density γ/sec·g	Source γ/sec	Source Density γ/sec·g	
				2.5 ft^3	55 gal
	❶	❷	❸	❹	❺
3.50	3.276E+10	3.529E+05	2.234E+11	2.385E+06	1.265E+06
2.75	4.746E+10	5.516E+05	3.009E+11	3.710E+06	1.970E+06
2.25	7.088E+10	8.933E+05	4.332E+11	5.806E+06	3.095E+06
1.83	1.143E+11	1.709E+06	7.317E+11	1.074E+07	5.761E+06
1.50	2.012E+11	4.030E+06	1.155E+12	2.416E+07	1.304E+07
1.17	4.023E+11	1.104E+07	1.933E+12	7.974E+07	5.255E+07
0.90	8.725E+11	3.475E+07	3.539E+12	2.563E+08	1.586E+08
0.70	1.859E+12	1.199E+08	6.238E+12	8.658E+08	5.086E+08
0.50	6.369E+12	8.392E+08	1.481E+13	5.893E+09	3.286E+09
Co-60	3.336E+11	8.082E+06	1.652E+12	5.789E+07	3.312E+07
Cs-137	2.362E+12	1.607E+08	7.150E+12	1.173E+09	6.674E+08

**For discrete source γ/sec · g limits, use column ❹ when the payload object meets the 2.5 ft^3 size requirements, or column ❺ when it meets the 55 gallon size requirements. When the size meets neither criterion use column ❷ for the γ/sec · g limits.*

Qualification Procedure

The Payload Qualification Flowchart (Figure 1) provides a graphical overview of the qualification process. The procedure below provides more detailed step-wise instructions.

1. Determine the number of types of material in the payload. For each type of material, determine the configuration (i.e., general or discrete), isotopic activity (in both γ/sec and γ/sec·g), dimensions, volume, and mass. Determine the payload totals for each parameter.

2. For payloads that include beta source(s) with maximum beta energies > 0.3 MeV and $\sum S_{\beta} \geq$ 2E+12 β/sec, convert each beta source to an equivalent gamma source. The equivalent gamma source, S_{γ}, equals 3.5E-04 S_{β} Z_w $E_{\beta avg}$ in photons per sec; where S_{β} is the beta source in β/sec , Z_w is the weighted average Z of the beta absorbing material, and $E_{\beta avg}$ is the average energy of the beta in MeV. The resulting equivalent photon source has strength S_{γ} at an energy of $E_{\beta max}$, the maximum beta energy. Include the equivalent photon source along with the other photon sources determined in Step 3.

3. Calculate the total γ/sec and total γ/sec·g for each photon energy of each payload item, ignoring photon energies below 0.3 MeV. ^{60}Co and ^{137}Cs may be treated like single "energies" since they have their own limits in Table 1. The energies in Table 1 are upper bounds, so photon energies must be rounded up. If any photons have energies above 3.5 MeV, the material is unacceptable for transport in the cask. For payloads with a large number of gammas, the gammas may be grouped into the energy groups in Table 1 and the photons sources determined for the group. Calculations of γ/sec·g should not include the mass of liners or other secondary containers.

4. For each payload item, select the two appropriate limit columns (❶ through ❺) in Table 1: one each for γ/sec and γ/sec·g. Base the γ/sec on the total γ/sec for the item, and the γ/sec·g on the highest-activity ("hottest") portions surveyed.

5. For each energy, calculate the γ/sec and γ/sec·g fractions (i.e., payload source/limit fraction). Select the smallest of each pair of fractions at each energy and add the resulting fraction to the running sum of fractions.

6. Repeat Steps 4-5 for each payload item.

7. If the running sum is less than 0.95, the payload's radiological source is acceptable.

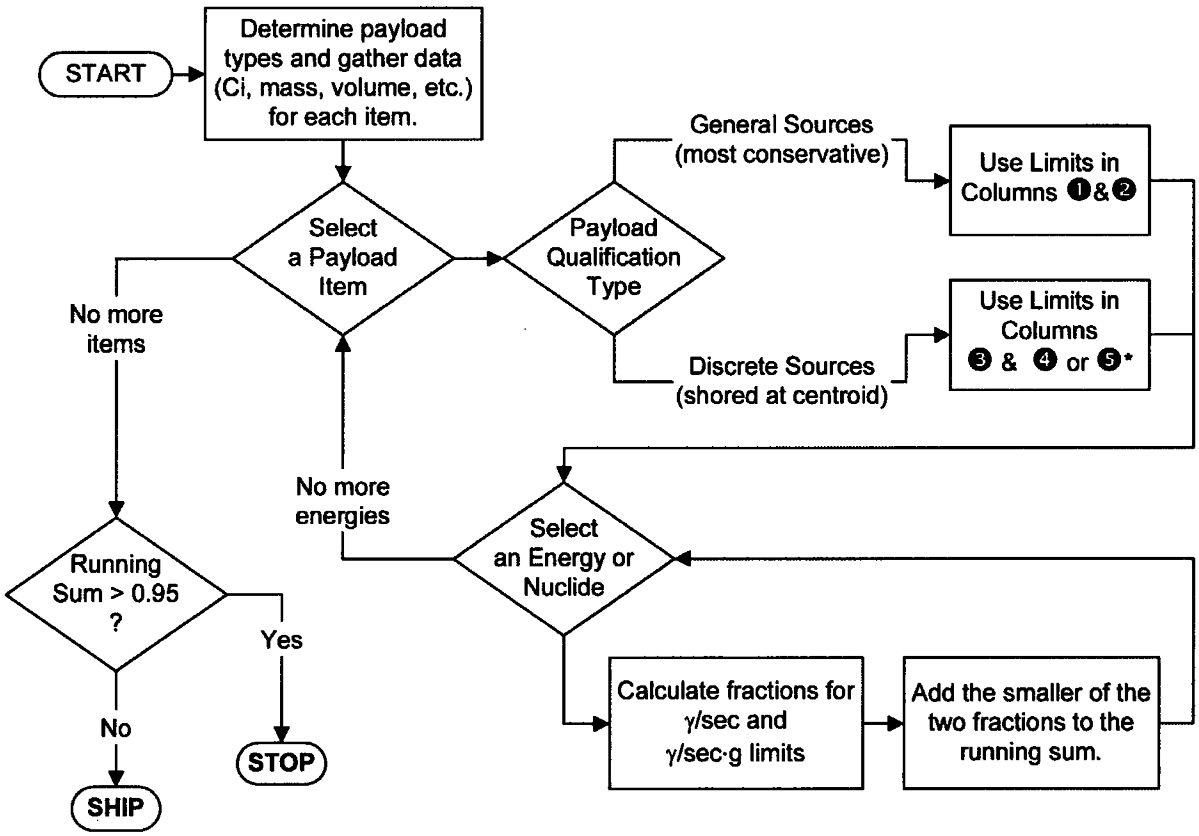


Figure 1 – Payload Qualification Flow Chart

<u>Example 1 - Cs-137 Source Capsule</u>

Problem: Determine the acceptability of a 50 Ci ^{137}Cs source to be centrally shored. The source is a metal capsule 2 cm in diameter by 10 cm long, and the Cs source pellet weighs 50 g.

Step 1: Characterize Source
Given in the problem statement.

Step 2: Convert Beta Source to Equivalent Gamma Source
Not applicable.

Step 3: Calculate Gamma Source Strengths and Source Densities
The qualification table has specific limits for ^{137}Cs, so it is not necessary to do the qualification by energy line. The source's Ci activity must be converted to γ/sec and γ/sec·g in order to calculate the source/limit fractions. ^{137}Cs produces 0.85 gammas per decay with an energy of 0.66 MeV. The total source is

$$3.7\times10^{10}\frac{d}{Ci}\times\frac{0.85\gamma}{d}\times50Ci=1.57\times10^{12}\frac{\gamma}{\sec},$$

and, dividing by 50 g, the total source density is 3.14E10 γ/sec·g.

Step 4: Select the Limits

Since this payload is to be shipped in a shored configuration, the payload is a "discrete" type payload. The size fits within the defined envelope for the 2.5 ft^3 payload, therefore the column ❸ and ❹ limits apply for γ/sec and γ/sec·g, respectively.

Steps 5-7 Sum the Fractions

For this example, there is only one fraction to calculate.

Type	Shape (Discrete Only)	Energy (MeV),or Nuclide	Payload Source Term		Limits			Fractions, F		F_{min}
			γ/sec	γ/sec·g	Energy	γ/sec	γ/sec·g	γ/sec	γ/sec·g	
Discrete	2.5 ft3	Cs-137	1.57E+12	3.15E+10	Cs-137	❸ 7.15E+12	❹ 1.17E+09	2.20E-01	2.68E+01	2.20E-01
									Sum:	2.20E-01

Since the sum is less than 0.95, the source is an acceptable payload.

Example 2 – Solidified Process Waste

Problem: Determine the acceptability of a 100 ft^3 secondary container containing solidified process waste. The activity is uniformly distributed. The measured weight of the filled container is 13,100 lbs, and the weight of the empty container is 1,100 lbs. The isotopic activity, determined by analysis of samples of the waste, is:
5 Ci of ^{60}Co, 10 Ci of ^{137}Cs, 50 Ci of ^{55}Fe, 4 Ci of ^{54}Mn, and 20 Ci of ^{90}Sr

Step 1: Characterize Source
Given in the problem statement.

Step 2: Convert Beta Source to Equivalent Gamma Source
^{90}Sr emits beta radiation through its own decay, plus the decay of its short-lived daughter product, ^{90}Y. So the beta production rate is 20 Ci * 3.7E+10 d/Ci *2 = 1.5E+12 β/sec. Since this is below the threshold of 2E+12 β/sec, the beta production is not significant and can be disregarded.

Step 3: Calculate Gamma Source Strengths and Source Densities
The qualification table has specific limits for ^{60}Co and ^{137}Cs, but it will be necessary to do the qualification by energy line for the remaining nuclides. After converting the Ci data to gamma energy lines for the remaining nuclides (neglecting any gamma energy lines < 0.3 MeV), the following source data are to be used for qualification. The γ/sec·g source densities are based on 12,000 lbs, the actual weight of the radioactive material.

Energy (MeV),or Nuclide	Payload Source Term	
	γ/sec	γ/sec·g
Co-60	3.70E+11	6.80E+04
Cs-137	3.15E+11	5.78E+04
0.8348	1.48E+11	2.72E+04

Step 4: Select the Limits
Since this payload does not meet the definition of either of the two discrete shored configurations (2.5 ft^3 or 55 gal), it is a "general" type payload. The limits in columns ❶ and ❷ apply for γ/sec and γ/sec·g, respectively.

Steps 5-7 Sum the Fractions
For this example, there are three lines: a ^{60}Co line, ^{137}Cs line, and one energy line representing ^{54}Mn (^{55}Fe and ^{90}Sr are disregarded because ^{55}Fe gammas are below 0.3 MeV, and the ^{90}Sr betas are below 2E+12 β/sec).

Line	Payload Item	Type	Shape (Discrete Only)	Energy (MeV),or Nuclide	Payload Source Term		Limits					Fractions, F		F_{min}
					γ/sec	γ/sec·g	Energy	γ/sec		γ/sec·g		γ/sec	γ/sec·g	
1	Solidified Waste Cont.	General		Co-60	3.70E+11	6.80E+04	Co-60	❶	3.34E+11	❷	8.08E+06	1.11E+00	8.41E-03	8.41E-03
2	Solidified Waste Cont.	General		Cs-137	3.15E+11	5.78E+04	Cs-137	❶	2.36E+12	❷	1.61E+08	1.33E-01	3.60E-04	3.60E-04
3	Solidified Waste Cont.	General		0.8348	1.48E+11	2.72E+04	0.9	❶	8.73E+11	❷	3.47E+07	1.70E-01	7.83E-04	7.83E-04
													Sum:	9.55E-03

Since the sum is less than 0.95, the container is an acceptable payload.

Example 3 – Dewatered Resin Liner

Problem: Determine the acceptability of a 100 ft^3 steel secondary container containing dewatered resin. The activity is uniformly distributed. The measured weight of the filled container is 13,100 lbs; the weight of the empty container is 1,100 lbs. The isotopic activity, determined by analysis of samples of the waste, is: 5 Ci of ^{60}Co, 10 Ci of ^{137}Cs, 50 Ci of ^{55}Fe, 4 Ci of ^{54}Mn, and 30 Ci of ^{90}Sr. Also included is a 100 cc piece of activated metal, not shored, with an activity of 0.5 Ci of ^{60}Co. The activated metal is steel with a density of 8 g/cm^3.
This differs from Example 2 in that there is more ^{90}Sr, and there is the additional piece of activated metal.

Step 1: Characterize Source
Given in the problem statement.

Step 2: Convert Beta Source to Equivalent Gamma Source
^{90}Sr emits beta radiation through its own decay, plus the decay of its short-lived daughter product, ^{90}Y. So the total beta production rate is 30 Ci * 3.7E+10 d/Ci * 2 = 2.22E+12 betas/sec. Since this is above the threshold of 2E+12 betas/sec, the beta production must be considered. Using the procedure to convert beta into equivalent gamma radiation described in Attachment 1, the $^{90}Sr/^{90}Y$ betas[1] will be treated as follows:

$E_{maxSr} = 0.54$ MeV, $E_{avgSr} = 0.19$ MeV
$E_{maxY} = 2.27$ MeV, $E_{avgY} = 0.93$ MeV
$Z_{Resin} = 5.6$, $Z_{Steel} = 26$
$Z_w = (12{,}000/13{,}100) \times 5.6 + (1{,}100/13{,}100) \times 26 = 7.3$
$S_{\gamma Sr} = (1.11E+12)(3.5E-04)(7.3)(0.19)$ = <u>5.39E+08 γ/s @ 0.54 MeV</u>
$S_{\gamma Y} = (1.11E+12)(3.5E-04)(7.3)(0.93)$ = <u>2.64E+09 γ/s @ 2.27 MeV</u>

Step 3: Calculate Gamma Source Strengths and Source Densities
This payload must be broken into two payload items, due to the physical and radiological differences between the resins and the activated metal.

<u>Resin Payload Item</u>

Like Example 2, the following source data are to be used for qualification of the gamma emitters.

Energy (MeV),or Nuclide	Payload Source Term	
	γ/sec·g	γ/sec·g
Co-60	3.70E+11	6.80E+04
Cs-137	3.15E+11	5.78E+04
0.8348	1.48E+11	2.72E+04

<u>Activated Metal Payload Item</u>

^{60}Co emits two photons per disintegration, therefore the total source for the activated metal is (0.5 Ci)(2 γ/d)(3.7E+10 d/sec-Ci) = 3.7E+10 γ/sec. Dividing by the mass of 100 g, the source density is 3.7E+08 γ/sec.

Step 4: Select the Limits

[1] Cember, H., "Introduction to Health Physics," Pergamon Press, 2nd Ed.

Resin Payload Item - Since this payload item does not meet the definitions of either of the two discrete shored configurations (2.5 ft^3 or 55 gal), it is a "general" type payload. The limits in columns ❶ and ❷ apply for γ/sec and γ/sec·g, respectively.

Activated Metal Payload Item – This payload item is small and fits within the defined envelope for the 2.5 ft^3 payload, however it is not shored, and so the activated metal is also a "general" type payload item. Columns ❶ and ❷ apply for the γ/sec and γ/sec·g limits, respectively.

Steps 5-7 Sum the Fractions

For this example, there are six lines: 1-3 are for the resin gamma emitters, 4-5 are for the bremsstrahlung gammas produced by ^{90}Sr and ^{90}Y, and one line for the activated metal ^{60}Co. The two ^{60}Co lines were kept separate for clarity, but could have been combined.

Line	Payload Item	Type	Shape (Discrete Only)	Energy (MeV),or Nuclide	Payload Source Term		Limits					Fractions, F		F_{min}
					γ/sec	γ/sec·g	Energy	γ/sec		γ/sec·g		γ/sec	γ/sec·g	
1	Resin	General		Co-60	3.70E+11	6.80E+04	Co-60	❶	3.34E+11	❷	8.08E+06	1.11E+00	8.41E-03	8.41E-03
2	Resin	General		Cs-137	3.15E+11	5.78E+04	Cs-137	❶	2.36E+12	❷	1.61E+08	1.33E-01	3.60E-04	3.60E-04
3	Resin	General		0.8348	1.48E+11	2.72E+04	0.9	❶	8.73E+11	❷	3.47E+07	1.70E-01	7.83E-04	7.83E-04
4	Resin (betas)	General		0.54	5.39E+08	9.90E+01	0.7	❶	1.86E+12	❷	1.20E+08	2.90E-04	8.26E-07	8.26E-07
5	Resin (betas)	General		2.27	2.64E+09	4.85E+02	2.75	❶	4.75E+10	❷	5.52E+05	5.56E-02	8.79E-04	8.79E-04
6	Metal	General		Co-60	3.70E+10	3.70E+08	Co-60	❶	3.34E+11	❷	8.08E+06	1.11E-01	4.58E+01	1.11E-01
													Sum:	1.21E-01

Since the sum is less than 0.95, the container is an acceptable payload.

Example 4 – Activated Waste with Non-Fixed Contamination

Problem: Determine the acceptability of a 100 ft^3 steel secondary container containing activated metal. The measured weight of the filled container is 7,100 lbs; the weight of the empty container is 1,100 lbs. The metal is composed of mildly activated steel, with non-fixed surface contamination. The contaminated surface area is estimated to be 500 ft^2. There is one small piece of activated steel with a significantly higher activity. Determine whether this smaller item can be included in the shipment, and whether it needs to be shored. The isotopic activities, determined by analysis of samples of the waste, are as follows:

- Most of the steel has similar radiological properties. Based on an analysis of the highest-activity sample, the constituents are: 20 Ci of ^{58}Co, 30 Ci of ^{60}Co, and 20 Ci of ^{54}Mn.
- The small activated metal item has a volume of 100 cc, dimensions of 1" x 1" x 24", with an activity of 6 Ci of ^{60}Co.
- The non-fixed crud contamination level, based on the highest-activity sample, is 50,000 dpm, which has been determined to be 50% ^{55}Fe, 30% ^{137}Cs, and 20% ^{60}Co. The contaminated surface area is 500 ft^2.

Step 1: Characterize Source
Given in the problem statement.

Step 2: Convert Beta Source to Equivalent Gamma Source
Not applicable since the beta source is less than 2E+12 β/sec.

Step 3: Calculate Gamma Source Strengths and Source Densities

100g Activated Metal Payload Item

^{60}Co emits two photons per disintegration, therefore the total source for the small activated metal item is (6 Ci)(2 γ/d)(3.7E+10 d/sec-Ci) = 4.44E+11 γ/sec. Dividing by the mass of 100 g, the source density is 4.44E+09 γ/sec.

Energy (MeV),or Nuclide	Payload Source Term	
	γ/sec	γ/sec·g
Co-60	4.44E+11	4.44E+09

Remaining Activated Metal Payload Item

^{60}Co emits two photons per disintegration, therefore the total ^{60}Co source for the activated metal is (30 Ci)(2 γ/d)(3.7E+10 d/sec-Ci) = 2.22E+12 γ/sec. The remaining nuclides, ^{58}Co and ^{54}Mn, were converted to individual energy lines[2] (E<0.3 MeV were neglected). Sources were divided by 2.72E+06 g (i.e., 6,000 lb) to obtain the γ/sec·g. The resulting sources are:

[2] MicroShield, Version 8.01, Grove Engineering.

Energy (MeV),or Nuclide	Payload Source Term	
	γ/sec	γ/sec·g
Co-60	2.22E+12	8.16E+05
0.511	2.21E+11	8.12E+04
0.8108	7.36E+11	2.70E+05
0.8348	7.40E+11	2.72E+05
0.8639	5.45E+09	2.00E+03
1.6747	3.97E+09	1.46E+03

Crud Payload Item

50,000 dpm is equivalent to 2.25E-08 Ci per 100 cm^2. The total activity is therefore (2.25E-08 Ci/100cm^2) (500 ft^2)(929 cm^2/ft^2) = 1.05E-04 Ci. The nuclide breakdown is therefore: 5.23E-05 Ci of ^{55}Fe, 3.14E-05 Ci of ^{137}Cs, and 2.09E-05 Ci of ^{60}Co. ^{55}Fe can be neglected since it does not emit any gammas > 0.3 MeV. The mass of the crud is not known, thus we can only use the γ/sec limit for qualification. If the mass is known, a γ/sec·g term can be determined. The source inputs are therefore:

Energy (MeV),or Nuclide	Payload Source Term	
	γ/sec	γ/sec·g
Co-60	1.55E+06	
Cs-137	9.88E+05	

Step 4: Select the Limits

If the 100g activated item requires shoring, it would meet the size criteria for the 55-gallon discrete shored configuration, so its limits would be columns ❸ and ❺ for γ/sec and γ/sec·g, respectively. Otherwise, since it would be unshored, the limits in columns ❶ and ❷ would apply for γ/sec and γ/sec·g, respectively. The remaining activated metal does not meet the definitions of either of the two discrete shored configurations (2.5 ft^3 or 55 gal), so it is a "general" type payload item. The limits in columns ❶ and ❷ apply for γ/sec and γ/sec·g, respectively. The crud is free to move within the cavity and is therefore a "general" type payload item. The limits in columns ❶ and ❷ apply for γ/sec and γ/sec·g, respectively.

Steps 5-7 Sum the Fractions

First we will try qualifying the payload without shoring the small activated item. Note that it is not acceptable to average the activated metal together with the small 100 g item.

Line	Payload Item	Type	Shape (Discrete Only)	Energy (MeV),or Nuclide	Payload Source Term γ/sec	Payload Source Term γ/sec·g	Limits Energy	Limits γ/sec	Limits γ/sec·g	Fractions, F γ/sec	Fractions, F γ/sec·g	F_{min}
1	100g activated item	General		Co-60	4.44E+11	4.44E+09	Co-60	❶ 3.34E+11	❷ 8.08E+06	1.33E+00	5.49E+02	1.33E+00
2	Remaining metal	General		Co-60	2.22E+12	8.16E+05	Co-60	❶ 3.34E+11	❷ 8.08E+06	6.65E+00	1.01E-01	1.01E-01
3	Remaining metal	General		0.511	2.21E+11	8.12E+04	0.7	❶ 1.86E+12	❷ 1.20E+08	1.19E-01	6.77E-04	6.77E-04
4	Remaining metal	General		0.8108	7.36E+11	2.70E+05	0.9	❶ 8.73E+11	❷ 3.47E+07	8.43E-01	7.78E-03	7.78E-03
5	Remaining metal	General		0.8348	7.40E+11	2.72E+05	0.9	❶ 8.73E+11	❷ 3.47E+07	8.48E-01	7.82E-03	7.82E-03
6	Remaining metal	General		0.8639	5.45E+09	2.00E+03	0.9	❶ 8.73E+11	❷ 3.47E+07	6.24E-03	5.76E-05	5.76E-05
7	Remaining metal	General		1.6747	3.97E+09	1.46E+03	1.83	❶ 1.14E+11	❷ 1.71E+06	3.47E-02	8.54E-04	8.54E-04
8	Crud	General		Co-60	1.55E+06		Co-60	❶ 3.34E+11	❷ 8.08E+06	4.64E-06		4.64E-06
9	Crud	General		Cs-137	9.88E+05		Cs-137	❶ 2.36E+12	❷ 1.61E+08	4.18E-07		4.18E-07
											Sum:	1.45E+00

This approach does not pass. Since the discrete shored payload items have higher limits, we can try to see if shoring the 100g item will pass.

Line	Payload Item	Type	Shape (Discrete Only)	Energy (MeV),or Nuclide	Payload Source Term γ/sec	Payload Source Term γ/sec·g	Limits Energy	Limits γ/sec	Limits γ/sec·g	Fractions, F γ/sec	Fractions, F γ/sec·g	F_{min}
1	100g activated item	Discrete	55 gal	Co-60	4.44E+11	4.44E+09	Co-60	❸ 1.65E+12	❺ 3.31E+07	2.69E-01	1.34E+02	2.69E-01
2	Remaining metal	General		Co-60	2.22E+12	8.16E+05	Co-60	❶ 3.34E+11	❷ 8.08E+06	6.65E+00	1.01E-01	1.01E-01
3	Remaining metal	General		0.511	2.21E+11	8.12E+04	0.7	❶ 1.86E+12	❷ 1.20E+08	1.19E-01	6.77E-04	6.77E-04
4	Remaining metal	General		0.8108	7.36E+11	2.70E+05	0.9	❶ 8.73E+11	❷ 3.47E+07	8.43E-01	7.78E-03	7.78E-03
5	Remaining metal	General		0.8348	7.40E+11	2.72E+05	0.9	❶ 8.73E+11	❷ 3.47E+07	8.48E-01	7.82E-03	7.82E-03
6	Remaining metal	General		0.8639	5.45E+09	2.00E+03	0.9	❶ 8.73E+11	❷ 3.47E+07	6.24E-03	5.76E-05	5.76E-05
7	Remaining metal	General		1.6747	3.97E+09	1.46E+03	1.83	❶ 1.14E+11	❷ 1.71E+06	3.47E-02	8.54E-04	8.54E-04
8	Crud	General		Co-60	1.55E+06		Co-60	❶ 3.34E+11	❷ 8.08E+06	4.64E-06		4.64E-06
9	Crud	General		Cs-137	9.88E+05		Cs-137	❶ 2.36E+12	❷ 1.61E+08	4.18E-07		4.18E-07
											Sum:	3.87E-01

Since the sum is less than 0.95, the container is an acceptable payload if the 100g item is shored at the centroid of the cavity.

Example 5 – Contaminated Soil

Problem: Determine the acceptability of a 100 ft^3 steel secondary container containing a contaminated soil mixture. The activity is not uniformly distributed. The measured weight of the filled container is 10,100 lbs; the weight of the empty container is 1,100 lbs. 5% of the payload mass is made up of small bits of grout used to immobilize contamination. The size of the grout chunks ranges from 0.1 cm to 10 cm. The grout contains ^{137}Cs at a maximum concentration of 215 Ci/ft^3. The remaining 95% of the material is soil with an activity of 10 Ci/ft^3 of ^{137}Cs. The density of the soil and grout are both 100 lb/ft^3. Activities were determined by analysis of samples of the most active representative waste.

Step 1: Characterize Source

Given in the problem statement.

Step 2: Convert Beta Source to Equivalent Gamma Source

Not applicable.

Step 3: Calculate Gamma Source Strengths and Source Densities

We will evaluate the payload two ways: one treating the entire payload as a single item with a bounding activity (γ/sec) and activity density (γ/sec·g), and the second assuming we will treat the payload as two separate items: grout and soil.

Grout Payload Item

The grout gamma source is (215 Ci/ft^3)(1 ft^3/100 lb)(9,000 lb*0.05)(3.7E+10 d/sec-Ci)(0.85 γ/d) = 3.04E+13 γ/sec. Dividing by the mass (450 lb, or 2.04E+05 g), the source density would be 1.49E+08 γ/sec·g.

Energy (MeV),or Nuclide	Payload Source Term	
	γ/sec	γ/sec·g
Cs-137	3.04E+13	1.49E+08

Soil Payload Item

The soil gamma source is (10 Ci/ft^3)(1 ft^3/100 lb)(9,000 lb*0.95)(3.7E+10 d/sec-Ci)(0.85 γ/d) = 2.69E+13 γ/sec. Dividing by the mass (8550 lb, or 3.88E+06 g), the source density would be 6.93E+06 γ/sec·g.

Energy (MeV),or Nuclide	Payload Source Term	
	γ/sec	γ/sec·g
Cs-137	2.69E+13	6.93E+06

Combined Grout/Soil Payload Item

If the payload is treated as a single item, the γ/sec is set equal to the sum of the γ/sec for both the grout and soil components. The γ/sec·g is set equal to that of the "hottest" component (i.e., the grout). Thus, the gamma source would be 5.73E+13 γ/sec (3.04E+13 + 2.69E+13). The γ/sec·g equals the 1.49E+08 value that applies for the grout.

Energy (MeV), or Nuclide	Payload Source Term	
	γ/sec	γ/sec·g
Cs-137	5.73E+13	1.49E+08

Step 4: Select the Limits

Since none of these payload items meets the definition of either of the two discrete shored configurations (2.5 ft^3 or 55 gal), they are "general" type payload items. The limits in columns ❶ and ❷ apply for γ/sec and γ/sec·g, respectively.

Steps 5-7 Sum the Fractions

As a first try, we attempt to qualify the payload as being two components: the grout and soil.

Line	Payload Item	Type	Shape (Discrete Only)	Energy (MeV),or Nuclide	Payload Source Term		Limits			Fractions, F		F_{min}
					γ/sec	γ/sec·g	Energy	γ/sec	γ/sec·g	γ/sec	γ/sec·g	
1	Grout	General		Cs-137	3.04E+13	1.49E+08	Cs-137	❶ 2.36E+12	❷ 1.61E+08	1.29E+01	9.28E-01	9.28E-01
2	Soil	General		Cs-137	2.69E+13	6.93E+06	Cs-137	❶ 2.36E+12	❷ 1.61E+08	1.14E+01	4.32E-02	4.32E-02
											Sum:	9.71E-01

Since the sum is greater than 0.95, the container is not an acceptable payload. It is acceptable, however, to treat the payload as a single (combined) item, with a γ/sec equal to the sum of the component (grout and soil) γ/sec values, and a γ/sec·g equal to that of the "hottest" component (i.e., the grout).

Line	Payload Item	Type	Shape (Discrete Only)	Energy (MeV),or Nuclide	Payload Source Term		Limits			Fractions, F		F_{min}
					γ/sec	γ/secž·g	Energy	γ/sec	γ/secž·g	γ/sec	γ/secž·g	
1	All-grout	General		Cs-137	5.73E+13	1.49E+08	Cs-137	❶ 2.36E+12	❷ 1.61E+08	2.43E+01	9.28E-01	9.28E-01
											Sum:	9.28E-01

Since the sum is less than 0.95, the container is an acceptable payload.

Attachment 2
Determination of Hydrogen Concentration

1. Determine the radionuclide concentration in the contents.
   For any package containing materials with radioactivity concentration not exceeding that for LSA, ensure the shipment occurs within 10 days of preparation, or within 10 days of venting the secondary container.
   For packages which satisfy the previous conditions, go to step 11, otherwise continue with step 2.

2. Determine the secondary package(s) void volume and the cask cavity void volume.

3. Identify the secondary container(s) vent path, if applicable

4. Determine the quantity of hydrogenous contents

5. Determine the G value of the hydrogenous contents per NUREG/CR-6673[3], Section 3.

6. Determine the energy deposition rate in the hydrogenous contents

7. Determine the hydrogen generation rate per NUREG/CR-6673, Section 4.2

8. Determine the effective hydrogen transport rate due to diffusion for the vent path; see NUREG/CR-6673, Section 4.1

9. Determine the shipping time to reach a hydrogen concentration of 5% in the package; see NUREG/CR-6673, Section 4.2.2.1 and Appendix F, Example #4.

10. If the time to reach 5% concentration is more than double the expected shipping time, the shipment meets the hydrogen concentration requirement.

11. Authorize the shipment

3

B. L. Anderson et al. *Hydrogen Generation in TRU Waste Transportation Packages* , NUREG/CR-6673, Lawrence Livermore National Laboratory, Livermore, CA, February 2000

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Acceptance tests for Configurations 1 and 2 have different weld examination and leak tests than Configuration 3. Maintenance is the same for all configurations.

8.1 ACCEPTANCE TESTS – CONFIGURATIONS 1 AND 2 (CASKS FABRICATED BEFORE APRIL 1, 1999)

Prior to the first use of the 8-120B package fabricated to Configuration 1 or 2, the following tests and evaluations will be performed.

8.1.1 Visual Examination

The package will be examined visually for any adverse conditions in materials or fabrication. Welds shall be examined for compliance to the drawings. Weld integrity shall be verified by visual examination and magnetic particle or dye penetrant. NDE examinations shall be performed by an ASME Certified inspector. Acceptance criteria for NDE shall be according to ASME Code Section III, Div. 1-Section NB5342 or NB5352 as applicable.

8.1.2 Structural Tests

No structural testing is required.

8.1.3 Leak Tests

This test shall be performed prior to acceptance and operation of a newly fabricated package in accordance with ASTM E-427 using a leak detector capable of detecting the applicable leak rates specified in Figures 4.4 and 4.7 in Chapter 4. Calibration of the leak detector shall be performed using a leak rate standard traceable to NIST. The standard's setting shall correspond to the approved leak rates specified in Figures 4.4 and 4.7 in Chapter 4.

All four containment boundary penetrations must be tested.

- The volume above the vent port Stat-O-Seal
- The volume between the drain line plug and interior of the cask
- The annulus between the o-ring seals of the primary lid
- The annulus between the o-ring seals of the secondary lid

All four of these volumes must be evacuated to a minimum vacuum of 20" Hg, and then be pressurized to a minimum pressure of 25 psig with pure dichlorodifluoromethane (R-12) or 1,1,1,2 – tetrafluoroethane (R-134a). Use the detector probe to "sniff" the following areas:

- The vent port penetration on the underside of the primary lid
- Around the head of the cap screw that plugs the drain line
- Interior side of the inner o-ring for the primary lid
- Interior side of the inner o-ring for the secondary lid

Leak detection shall be in accordance with the specifications of ASTM E-427.

Any condition, which results in leakage in excess of the applicable values specified in Figures 4.3 and 4.6 in Chapter 4 shall be corrected.

8.1.4 Component Tests

Gaskets and seals will be procured and examined in accordance with the Energy*Solutions* Quality Assurance Program.

8.1.5 Test for Shielding Integrity

Shielding integrity of the package will be verified by gamma scan or gamma probe methods to assure the package is free of significant voids in the poured lead shield annulus. All gamma scanning will be performed on a 4-inch square or less grid system. The acceptance criteria will be that voids resulting in shield loss in excess of 10 % of the normal lead thickness in the direction measured shall not be acceptable.

8.1.6 Thermal Acceptance Tests

No thermal acceptance testing will be performed on the 8-120B package. Refer to the Thermal Evaluation, Chapter 3.0 of the report.

8.2 ACCEPTANCE TESTS – CONFIGURATION 3 (CASKS FABRICATED AFTER APRIL 1, 1999)

Prior to the first use of an 8-120B package fabricated to Configuration 3, the following tests and evaluations will be performed:

8.2.1 Visual Inspections and Measurements

Throughout the fabrication process, confirmation by visual examination and measurement are required to be performed to verify that the 8-120B packaging dimensionally conforms to the drawing referenced in the current Certificate of Compliance for the 8-120B.

The packaging is also required to be visually examined for any adverse conditions in materials or fabrication that would not allow the packaging to be assembled and operated per Section 7.0 or tested in accordance with the requirements of Section 8.0.

Throughout the fabrication process, the fabricator shall request approval from Energy*Solutions* prior to implementation of any options allowed in the drawing.

8.2.2 Weld Examinations

8.2.2.1 Containment boundary welds identified on drawing C-110-E-0007 are required to be inspected and are required to meet the acceptance requirements of ASME Code, Section III, Division I, Subsection ND, Article ND-5000.

8.2.2.2 The Containment Boundary welds listed below shall be inspected by radiographic examination (RT) in accordance with ASME Code Section III – Division I, Subsection ND-5200, with weld acceptance per ASME Code, Section III, Division I, Subsection ND, Article ND-5320. In lieu of RT examination, the weld may be examined by ultrasonic examination (UT) plus magnetic particle examination (MT) per ASME Code, Section III, Division I, Subsection ND, Article ND-5330 and Article ND-5340 respectively only as permitted by ASME Code, Section III, Division I, Subsection ND, Article ND-5279.

On drawing C-110-E-0007, the welds to be examined by RT or UT+MT per above provisions are:

a. Weld between Item 3, Inner Cask Shell and Item 4, Bolting Ring.
b. Weld between Item 3, Inner Cask Shell and Item 5A, Cask Bottom Plate.
c. Any seam welds on Item 3, Inner Cask Shell.
d. Weld between Item 17, and Item 18, Primary Lid.

The Containment boundary welds listed below are required to be inspected by liquid penetrant examination (PT), as noted, and are required to meet the acceptance requirements of ASME Code, Section III, Division I, Subsection ND, Article ND-5350.

On drawing C-110-E-0007, the welds to be examined by PT are:

e. Weld between Item 9, Primary Lid Seal Seating Plate and Item 4, Bolting Ring.
f. Any seam welds on Item 9, Primary Lid Seal Seating Plate.
g. Weld between Item 21, O-Ring Seal Plate and Item 17, Primary Lid.
h. Weld between Item 21, O-Ring Seal Plate and Item 36, Secondary Lid.
i. Weld between Item 19, Secondary Lid Seal Seating plate and Item 18, Primary Lid.
j. Weld between Item 19 and Item 20 Secondary Lid Seal Seating Area.
k. Any seam welds on Items 19 or 20 Secondary Lid Seal Seating Area.

8.2.2.3 Non-containment boundary welds on drawing C-110-E-0007 are required to be inspected and are required to meet the acceptance requirements of ASME Code, Section III, Division I, Subsection NF, Article NF-5000.

Non-containment boundary welds identified on drawing C-110-E-0007 as MT are required to be inspected in accordance with ASME Code Section III – Division I, Subsection NF-5230, and are required to meet the acceptance requirements of ASME Code, Section III, Division I, Subsection NF, Article NF-5340.

On drawing C-110-E-0007, the welds to be examined by MT are:

a. Weld between Item 5, Cask Bottom Inner Plate and Item 3, Inner Cask Shell.
b. Weld between Item 5, Cask Bottom Inner Plate and Item 5A, Cask Bottom Plate.
c. Weld between Item 1, Outer Cask Shell and Item 4, Bolting Ring.
d. Weld between Item 36, and Item 37, Secondary Lid.
e. Any seam welds on Item 1, Outer Cask Shell.

8.2.2.4 The Non-containment boundary welds listed below are required to be inspected by magnetic particle examination (MT) in accordance with ASME Code Section III – Division I, Subsection NF-5230, after the root pass and after the cover pass and are required to meet the acceptance requirements of ASME Code, Section III, Division I, Subsection NF, Article NF-5340.

On drawing C-110-E-0007, the welds to be examined by MT (Root Pass) and MT (After Cover Pass) are:

a. Weld between Item 5B, Cask Bottom Plate and Item 1, Outer Cask Shell.
b. Weld between Item 5A and Item 5B, Cask Bottom Plate.

8.2.2.5 Welds on lifting and tiedown lugs identified on drawing C-110-E-0007 are required to be inspected by magnetic particle examination (MT) and are required to meet the acceptance requirements of ASME Code, Section III, Division I, Subsection ND, Article ND-5340 or NF, Article NF-5340. Inspection shall be before and after 150% load test.

8.2.3 Structural and Pressure Tests

A pressure test of the containment system will be performed as required by 10CFR71.85. As determined in Section 3.4.4, the maximum normal operating pressure for the cask cavity is 35 psig; therefore the minimum test pressure will be 1.5 x 35 = 52.5 psig. The hydrostatic test pressure will be held for a minimum of 10 minutes prior to initiation of any examinations. Following the 10 minute hold time, the cask body, lid and lid/body closure shall be examined for leakage. Any leaks, except from temporary connections, will be remedied and the test and inspection will be repeated. After depressurization and draining, the cask cavity and seal areas will be visually inspected for cracks and deformation. Any cracks or deformation will be remedied and the test and inspection will be repeated.

8.2.4 Leakage Tests

8.2.4.1 General requirements

- Testing method – Per ANSI N-14.5 in accordance with ASTM E-427 if using a halogen leak detector or ASTM E-499 if using a helium leak detector.
- Test Sensitivity – the test method must be capable of meeting the appropriate sensitivity requirements specified in Figures 4.4 or 4.7 in Section 4.0. Calibration of the leak detector shall be performed using a leak rate standard traceable to NIST.
- The leak standard's setting shall correspond to the approved leak test rate (see Section 4.0).
- Any condition, which results in leakage in excess of the maximum allowable leak rate specified in Figures 4.3 or 4.6 (depending on the test gas used), shall be corrected and re-tested.

8.2.4.2 Testing of the entire containment boundary will be performed prior to lead pour to allow access to all containment welds. The containment boundary includes: the inner shell, the cask bottom base plate (BOM 5A), the bolting ring, the lids, the O-ring seal plates of both lids, the inner O-ring of both lids, and the vent port cap screw and its seal.

- (Optional) Insert the sealed metal cavity filler canister into the cask cavity. Verify the canister does not obstruct the vent penetration. The metal must be chemically compatible with the cask liner and the test gas.
- Assemble the cask lids per Section 7.1.
- Evacuate the cask cavity to 20" Hg vacuum, minimum (sealed metal cavity filler canister may be used within the cask cavity)
- Pressurize the cask cavity to a minimum pressure of:
  1) 25 psig with pure 1,1,1,2 – tetrafluoroethane (R-134a),
  or
  2) 1 psig with pure helium.
- Check for leakage of the inner shell and base plate components
- Measure the leakage of the inner (containment) O-ring via the test port in each lid.
- Check for leakage at the vent port.

8.2.5 Component and Material Tests

Energy*Solutions* will apply its USNRC approved 10CFR71 Appendix B Quality Assurance Program, which implements a graded approach to quality based on a component's or material's importance to safety to assure all materials used to fabricate and maintain the 8-120B are procured with appropriate documentation which meet the appropriate tests and acceptance criteria for packaging materials.

This includes as example:
ASTM steel material used for shells, lids, bolts, etc. will comply with and meet ASTM manufacturing requirements.
Seals will meet requirements of ES-C-038, which is included in Appendix 8.3.1.
The impact limiter foam will meet the requirements of ES-M-175, which is included in Appendix 8.3.1.

8.2.6 Shielding Tests

Shielding integrity of the package will be verified by gamma scan to assure the package lead layer meets or exceeds the minimum thickness specified on the cask drawing. All gamma scanning will be performed on a 4-inch square or less grid system. The acceptance criteria (maximum value) will be determined by: Option 1) measurement of the maximum value using a test block, which has shield layers that replicate the cask geometry per the drawing, using the gamma scan source and reproducing the source/shield/detector geometry that will be used during the scan of the cask, or Option 2) calculation of the maximum value using detailed modeling software (MCNP or equivalent) incorporating the specific cask dimensions from the drawing and the source/shield/detector geometry applicable to the gamma scan. Any location on the cask which shows a gamma scan value greater than the maximum value will be identified as unacceptable. All unacceptable areas will be remedied and re-scanned.

8.2.7 Thermal Tests

No thermal acceptance testing will be performed on the 8-120B packaging. Refer to the Thermal Evaluation, Section 3.0 of this report.

8.2.8 Miscellaneous Tests

No miscellaneous testing will be performed on the 8-120B packaging.

8.3 MAINTENANCE PROGRAM

Energy*Solutions* operates an ongoing preventative maintenance program for all shipping packages. The 8-120B package will be subjected to routine and periodic inspection and tests as outlined in this section and the approved procedure based on these requirements. Defective items are replaced or remedied, including testing, as appropriate.

Examples of inspections performed prior to each use of the cask include:

Cask Seal Areas: O-rings are inspected for any cracks, tears, cuts, or discontinuities that may prevent the O-ring from sealing properly. O-ring seal seating surfaces are inspected to ensure they are free of scratches, gouges, nicks, cracks, etc. that may prevent the O-ring from sealing properly. Defective items are replaced or remedied, as appropriate and tested in accordance with Section 8.3.2.

Cask bolts, bolt holes, and washers are inspected for damaged threads, severe rusting or corrosion pitting. Defective items are replaced or remedied, as appropriate.

Lift Lugs and visible lift lug welds are inspected to verify that no deformation of the lift lug is evident and that no obvious defects are visible. Defective items are replaced or remedied, as appropriate and tested in accordance with Section 8.2.2.5.

8.3.1 Structural and Pressure Tests

No routine or periodic structural or pressure testing will be performed on the 8-120B packaging.

8.3.2 Leakage Tests

8.3.2.1 Periodic and Maintenance Leak Test.

The 8-120B packaging shall have been leak tested as described below within the preceding 12-month period before actual use for shipment and after maintenance, repair (such as weld repair), or replacement of components of the containment system.

The 8-120B packaging seals shall have been replaced within the 12-month period before actual use for shipment.

General requirements

- Testing method – Per ANSI N-14.5 in accordance with ASTM E-427 if using a halogen leak detector or ASTM E-499 if using a helium leak detector.
- Test Sensitivity – the test method must be capable of meeting the appropriate sensitivity requirements specified in Figures 4.4 or 4.7 or in Section 4.8. Calibration of the leak detector shall be performed using a leak rate standard traceable to NIST.
- The leak standard's setting shall correspond to the approved leak test rate (see Section 4.0).
- Any condition, which results in leakage in excess of the appropriate maximum allowable leak rate specified in Figures 4.3, 4.6 or Section 4.8, shall be corrected and re-tested.

Testing of the Lids and Vent

- (Optional) Insert the sealed metal cavity filler canister into the cask cavity. Verify the canister does not obstruct the vent penetration. The metal must be chemically compatible with the cask liner and the test gas.
- Assemble the cask lids per Section 7.1.
- Evacuate the cask cavity to 20" Hg vacuum (minimum) or 90% vacuum for the leak tight test.
- Pressurize the cask cavity to a minimum pressure of:
  1) 25 psig with pure 1,1,1,2 – tetrafluoroethane (R-134a),
     or
  2) 1 psig with pure helium.
- Measure the leakage of the inner (containment) O-ring via the test port in each lid.
- Measure the leakage of the vent port.

Testing of the Lids – Optional Method

- Assemble the cask lids per Section 7.1.
- Connect to the O-ring test port on the lid and evacuate the annulus between the cask lid O-rings to 20" Hg vacuum (minimum)

- Pressurize the O-ring annulus to a minimum pressure of 25 psig with pure 1,1,1,2 – tetrafluoroethane (R-134a),
- Check for leakage of the inner (containment) O-ring by moving a detector probe along the interior surface of the inner seal according to the specifications of ASTM E-427.

Testing of the Vent – Optional Method

- Assemble the cask Vent Port Cap Screw and Seal per Section 7.1.
- With the vent port cover (Item 30) removed, connect to and evacuate the volume above (lid exterior) the Vent Port Cap Screw and Seal (Items 26 and 27) to 20" Hg vacuum (minimum)
- Pressurize the volume to a minimum pressure of 25 psig with pure 1,1,1,2 – tetrafluoroethane (R-134a),
- Check for leakage of the Vent Port Cap Screw and Seal by moving a detector probe along the interior surface of the Primary Lid in the area of the vent port according to the specifications of ASTM E-427.

The requirements for Periodic and Maintenance Leak Testing of the 8-120B are summarized in Table 8.1.

Table 8.1
Periodic and Maintenance Leak Test of 8-120B

Component	Test Gas	Max. Leak Rate	Minimum Sensitivity	Test Pressure	Procedure	Alternate Procedure
Lid	R-134a	Fig. 4.3	Fig. 4.4	Evacuate cask cavity to 20" Hg then pressurize to 25 psig.	After pressurizing the cask cavity with the test gas, check for gas leakage from the cask Lid inner O-ring using the cask Lid test port.	After pressurizing between the lid O-ring annulus with the test gas, check for gas leakage from the cask Lid inner O-ring using a detector probe.
	Helium	Fig. 4.6	Fig. 4.7	Evacuate cask cavity to 20" Hg, or 90% vacuum for the leak tight test, then pressurize to 1 psig.	After pressurizing the cask cavity with the test gas, check for gas leakage from the cask Lid inner O-ring using the cask Lid test port.	N/A
Vent Port	R-134a	Fig. 4.3	Fig. 4.4	Evacuate cask cavity to 20" Hg then pressurize to 25 psig.	After pressurizing the cask cavity with the test gas, check for gas leakage from the Vent Port and Seal.	After pressurizing the volume above the Vent Port Cap Screw and Seal with the test gas, check for gas leakage from the vent penetration on the inner side of the lid using a detector probe.
	Helium	Fig. 4.6	Fig. 4.7	Evacuate cask cavity to 20" Hg, or 90% vacuum for the leak tight test, then pressurize to 1 psig.	After pressurizing the cask cavity with the test gas, check for gas leakage from the Vent Port Cap Screw and Seal.	N/A

8.3.2.2 Pre-Shipment Leak Test

a. This test is required before each shipment of Type B material quantities. The test will verify that the containment system has been assembled properly.

Note: The pre-shipment leak test is not required before a shipment if the contents meet the definition of low specific activity materials or surface contaminated objects in 10CFR71.4, and also meet the exemption standard for low specific activity materials or surface contaminated objects in 10CFR71.14(b)(3)(i).

b. The test will be performed by pressurizing the annulus between the O-ring seals of each lid, or inlet to the vent port with dry air or nitrogen.

Note: The pre-shipment leak test is typically performed using a test manifold that may be constructed from tubing, fittings, isolation valves and a pressure gauge. Any test apparatus used for this test must have an internal volume, with isolation valves closed and the apparatus connected to the test port location, of less than or equal to 10 cm^3 to achieve the required test sensitivity for the hold time specified in Section 8.3.2.2.d.

Note: If air is used for the test, the air supply should be clean and dry. If it is not, or if the quality of the air supply is uncertain, the test should be performed with nitrogen to ensure reliable results.

c. The test shall be performed using a pressure gauge, accurate within 1%, or less, of full scale.

d. The test pressure shall be applied for at least 15 minutes for the lid or vent port. A drop in pressure of greater than the minimum detectable amount shall be cause for test failure. The maximum sensitivity of the gauge shall be 0.1 psig.

e. Sensitivity at the test conditions is equivalent to the prescribed procedure sensitivity of 10^{-3} ref-cm^3/sec based on dry air at standard conditions as defined in ANSI N14.5-1997 (See Section 4.5 for the determination of the test conditions).

Table 8.2 summarizes pre-shipment leak test requirements for the 8-120B:

Table 8.2
Pre-Shipment Leak Test of 8-120B Components

Component	Hold Time	Procedure
Lid	15 min.	Connect test manifold to the test port. Pressurize void between O-rings with the test gas, close the isolation valves and hold for the minimum hold time. A drop in pressure of greater than the minimum detectable amount shall be cause for test failure.
Vent Port	15 min.	Remove the threaded cap covering the vent port. Connect test manifold to the vent port. Pressurize the seal and head of the vent port cap screw for the minimum hold time. A drop in pressure of greater than the minimum detectable amount shall be cause for test failure.

8.3.3 Component and Material Tests

Cask seals (O-rings) are inspected each time the cask lids or vent port cap screw are removed. Inspection and replacement of the seal is discussed in Section 8.3.

New seals are lightly coated with a lightweight lubricant such as Parker Super O-Lube or equivalent prior to installation. The lubricant will minimize deterioration or cracking of the elastomer during usage and tearing if removal from the dovetail groove is necessary for inspection. Coating the exposed surfaces of installed lid seals with the lightweight lubricant immediately prior to closing the lid can help to minimize deterioration or cracking of the seal during use. Excess lubricant should be wiped off before closing the lid.

Painted surfaces, identification markings, and match marks used for closure orientation shall be visually inspected to ensure that painted surfaces are in good condition, identification markings are legible, and that match marks used for closure orientation remain legible and are easy to identify.

Visible cask external and cavity welds shall be inspected within twelve months prior to use to verify that the welds specified by the applicable cask drawing are present and that no obvious weld defects are visible. If paint is covering these welds, the inspection may be completed without removing the paint.

8.3.4 Thermal Tests

No periodic or routine thermal testing will be performed on the 8-120B packaging.

8.3.5 Miscellaneous Tests

8.3.5.1 Repair of Bolt Holes

Threaded inserts may be used for repair of bolt holes. The following steps shall be performed for each repair using a threaded insert.

a. Install threaded insert(s), sized per manufacturer's recommendation, per the manufacturer's instructions.

b. At a minimum, each repaired bolt hole(s) will be tested for proper installation by assembling the joint components where the insert is used and tightening the bolts to their required torque value.

Note: If the repair is to bolt holes for lifting components, then a load test will also be performed to the affected components equal to 150% of maximum service load.

c. Each threaded insert shall be visually inspected after testing to insure that there is no visible damage or deformation to the insert.

8.3.6 APPENDICES

8.3.6.1 Appendix

Polyurethane Foam Specification ES-M-175
(available on request)

8-120B Seal Specification ES-C-038
(available on request)

Seal Specification for the 8-120B Cask

Revision 0

Authored By:	Michael Frassica, Project Engineer	2/17/12 Date
Reviewed By:	Phillip Thomas, Cask Operations Manager	2/17/12 Date
Approved By:	Patrick L. Paquin, Director - Products	2/17/12 Date

- [X] New
- [ ] Title Change
- [ ] Revision
- [ ] Rewrite
- [ ] Cancellation

Effective Date 2/21/12

TABLE OF CONTENTS

1 SCOPE

1.1 Purpose

This document provides the technical specification and performance tests for elastomeric compounds used as containment boundary seals for the Energy*Solutions* licensed 8-120B transport cask.

1.2 Introduction

The intention of this specification is to provide consistent elastomeric criteria across several types of compounds. Example compounds that are well suited in sealing applications are (but not limited to) silicone, butyl (IIR), and ethylene propylene (EPDM). To demonstrate containment boundary elastomer(s) ability to perform under Hypothetical Accident Conditions (HAC) and Normal Conditions to Transport (NCT), selected compound(s) shall be subjected to and successfully demonstrate the ability to pass tests and inspections communicated herein.

1.3 General Overview

Elastomeric seal material is a general term which comprises a wide variety of polymers having elastic properties similar to natural rubber. Therefore selected elastomeric seal material for containment boundaries will be delineated by meeting the requirements of this specification. The helium permeability requirement (Section 4.1.3) only applies to seals used in packages subject to leak tight requirements.

2 REFERENCES

2.1 ASTM D2240, Standard Test Method for Rubber Property; Durometer Hardness

2.2 ASTM D2137, Standard Test Methods for Rubber Property - Brittleness Point of Flexible Polymers and Coated Fabrics

2.3 Parker O-Ring Handbook, ORD 5700, Parker Seals, O-ring Division, 2007

2.4 Safety Analysis Report for Model 8-120B Type B Shipping Packaging (latest revision)

2.5 Fastener and Fitting Seals Catalog, CSS 5125, Parker Hannifin Corporation, Composite Sealing Systems Division

2.6 ASTM E1069-85, Standard Test Method for Testing Polymeric Seal Materials for Geothermal or High Temperature Service, or both, Under Sealing Stress

3 MATERIAL

3.1 Elastomeric Selection

Elastomeric compounds to be identified by Energy*Solutions* Q-Level I supplier on Approved Suppliers List (ASL) or by ASL laboratory utilizing fourier transform infrared spectrometry (FTIR).

3.2 Restrictions

Seals fabricated from fluoropolymers shall be excluded for containment seals on the 8-120B cask.

4 Physical Properties

4.1 Mechanical Properties for Elastomeric Seals

4.1.1 Durometer; compounds shall have a hardness between 50 and 70 Shore A, Reference 2.1.

4.1.2 Low Temperature Compatibility; compounds shall pass the low temperature brittleness test, Method A at -40°F, Reference 2.2.

4.1.3 Permeability (For Leak Tight packaging only); compounds shall have a maximum permeability for helium gas of 100 x 10^{-8}[$cm^3 \cdot cm / cm^2 \cdot sec \cdot bar$] at ambient temperature, Reference 2.3

4.2 Dimensional Conformance

4.2.1 Cross section of seal: O-rings' cross section as stated on the SAR bill of material. Manufactured seals (e.g. Parker Stat-O-Seal) shall conform to manufacturer literature.

4.2.2 Outside Diameter (Reference 2.4, Drawing C-110-E-0007):

Seal Position	BOM No.	Outside Dimension (in.)	Cross Sectional Diameter (in.)	Quality Level
Primary Inner O-ring	22	63"+/- 1/2"	0.285" +/- 0.010"	1
Primary Outer O-ring	23	65"+/- 1/2"	0.285" +/- 0.010"	3
Secondary Inner O-ring	24	30" +/- 1/2",	0.285" +/- 0.010"	1
Secondary Outer O-ring	25	32" +/- 1/2"	0.285" +/- 0.010"	3
Vent Bolt Seal	26	Reference 2.5	Reference 2.5	1

5 Conditional Testing

5.1 Temperature - Pressure Testing

5.1.1 Seal Test of Normal Condition of Transport (NCT)

Utilizing testing methods as outlined in Reference 2.6 with the noted alteration (§§5.1.1.1-5.1.1.3) to simulate the environment during the 8-120B NCT condition.

5.1.1.1 Test temperature shall be maximum allowable (-0, + 10%) from Table 3-1 of Ref. 2.4

5.1.1.2 Test pressure shall be MNOP (-0, + 10%) from Section 3.3.2 of Ref. 2.4

5.1.1.3 Test duration shall be a minimum of 1000 hrs

5.1.2 Seal Test of Hypothetical Accident Condition (HAC)

Utilizing testing methods as outlined in Reference 2.6 with the noted alteration (§§5.1.2.1 -5.1.2.3) to reproduce 8-120B SAR HAC conditions

5.1.2.1 Test temperature shall be maximum allowable (-0, + 10%) from Table 3-2 of Ref. 2.4

5.1.2.2 Test pressure shall be maximum HAC pressure (-0, + 10%) from Section 3.4.3 of Ref. 2.4

5.1.2.3 Test duration shall be minimum of 70 hrs

5.2 Considerations for Irradiation of Elastomeric Seals

Elastomers experience some degradation in elasticity when exposed to radiation levels of 10^7 rad. Using conservative factors, calculation for the 8-120B cask show a maximum annual dose approaching 10^6 rad. At these levels, typical elastomers physical properties do not degrade significantly, Reference 2.3. Maximum installation time of all 8-120B seals is 365 days. Therefore selection of elastomeric compounds will not be limited by the effects of radiation during installation except as noted in §3.2.

6 QUALITY ASSURANCE

The procurement of seals shall comply with Energy*Solutions* Quality Assurance Program ES-QA-PG-001 latest revision.

ENERGY*SOLUTIONS*

AFFIDAVIT
SUBMITTED BY ENERGYSOLUTIONS
CONCERNING CONFIDENTIAL INFORMATION AND TRADE SECRETS

STATE OF SOUTH CAROLINA]
] *ss.*
COUNTY OF LEXINGTON]

I, James R. Carlson II, depose and say that I am duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified below as proprietary. The following documents and corresponding data files, which are included with our submittal letter EL&P-016-12, Response to RAI for the 8-120B Package contains proprietary information that should be withheld from public disclosure:

Proprietary Calculations, which contains the following proprietary documents and corresponding data files:

NU-391, Rev.2: 8-120B Shielding Response

I have personal knowledge of the criteria and procedures utilized by Energy*Solutions* in designating information as a trade secret or as confidential information of a commercial or financial nature. These calculations contain unique information and methods that have been developed by the Energy*Solutions'* staff for the evaluation of transportation packaging. These methods are considered confidential information that includes Company trade secrets incorporated into such evaluation processes. The proprietary information submitted to the Commission contains the type of information Energy*Solutions* regards as protected and of the type not to be disclosed to unauthorized persons.

The information designated here as proprietary is not available from public sources. Public disclosure of this information would cause substantial harm to the competitive position of Energy*Solutions*. The Company has made substantial investments in salaries and capital equipment and has committed to refine and improve Energy*Solutions*' radioactive waste management system. Competitors of Energy*Solutions* would have great difficulty in duplicating the methods developed by Energy*Solutions*, due not only to the financial investment of Energy*Solutions*, but also to the unique skills, talents, and expertise of Energy*Solutions* employees who have developed these concepts. Disclosure of this information could cause Energy*Solutions* to lose the financial opportunity and business associated with this and other projects similar in nature.

James R Carlson II

James R. Carlson II
Global Commercial Group Application Engineering Manager (Technology, Engineering & Services)

STATE OF SOUTH CAROLINA]
] *ss.*
COUNTY OF LEXINGTON]

On this 15th day of May 2012, before me, a Notary Public in and for the State of South Carolina, duly commissioned and sworn, personally appeared James Carlson, Engineering Manager for Energy*Solutions*, and on oath stated that he was authorized to make this affidavit on behalf of the corporation.

IN WITNESS WHEREOF, I have set my hand and affixed my official seal the day and year first above written.

5/14/12
Notary Public, State of South Carolina

ENERGYSOLUTIONS

30 May 2012
EL&P-020-12

Mr. Pierre Saverot
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT:	Amendment Request for Certificate of Compliance No. 9168 for the Model 8-120B Package - Request for Additional Information
Docket No.	71-9168
TAC No.	L24549

Dear Mr. Saverot:

Energy*Solutions* provides the enclosed CD to be provided to NRC Document Control as an attachment to our response to the second request for additional information (RAI) dated 28 March, 2012.
Please forward the CD to DCD along with the paper copy of the submittal previously provided.
Should you have any additional questions about the response, please contact me at (803) 758-1898.

Sincerely,

Mark Whittaker
Sr. Health Physicist, Radiological Services

Enclosure:

- References, Documents, and Data (Proprietary and Non-proprietary, provided on CD/DVD)

Suite 100, Center Point II
100 Center Point Circle
Columbia, South Carolina 29210