Return to Florence Brown
396-55

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OCT 25 1985

Project No. M-40

Nuclear Assurance Corporation
ATTN: Colman B. Woodhall
Vice President
5720 Peachtree Parkway
Norcross, GA 30092

Gentlemen:

In response to your submittal, docketed on December, 19, 1984 under Project No. M-40, we have initially reviewed your Topical Report entitled, "Topical Safety Analysis Report for the NAC Storage/Transport Cask for Use at an Independent Spent Fuel Storage Installation." Our detailed comments are enclosed.

If you have any questions regarding our comments, we shall be happy to discuss them with you.

Sincerely,

Leland C. Rouse, Chief
Advanced Fuel and Spent Fuel
Licensing Branch
Division of Fuel Cycle and
Material Safety

Enclosure:
As stated

OFC:FCAF	:FCAF	:	:	:	:	:
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Comments Relating to the Topical SAR for the NAC Storage/Transport Cask

1.2 General Description of the Installation

1.2.1 Principal Design Criteria

It is assumed that the design surface dose rate is in terms of mrem/hr. This abbreviation should be used consistently throughout the TSAR rather than mr/hr.

2.0 Site Characteristics

2.1 Geography and Demography of Site Selected

2.1.2 Site Description

The reference to 10 CFR 71.68(b) should be 10 CFR 72.68(b). The limits stated in 10 CFR 72.67(a) and 10 CFR 72.68(b) are for direct radiation and effluent (airborne radioactivity) releases combined. The NAC statement on page 2.1-2 makes no mention of the effluent release component.

2.3 Meteorology

2.3.4 Diffusion Estimates

How was the atmospheric diffusion factor calculated? What assumptions, if any, were made in the computations? Give references.

2.7 Summary of Site Conditions Affecting Construction and Operating Requirements

The "distance from cask to boundary" in Table 2.7-1 (Bounding Site Characteristics) is equal to or greater than 100 meters. However, Section 7.4.3 (Occupational Dose) gives the minimum distance to the ISFSI site boundary for compliance with 10 CFR 72.67(a) as 187.2 meters for a single cask. Why is this distance not presented here?

3.0 Principal Design Criteria

3.1 Purpose of Installation

3.1.1 Materials to be Stored

a. Was ORIGIN used to determine the gamma and neutron source strengths and fission product gas inventory? The code should be identified along with any relevant irradiation conditions. Furthermore, output results should be appended.

b. Why are the gamma results in Table 3.1-3 (Photon Spectrum for Design Basis Fuel) given per cm of active fuel length and not per assembly? Is the total source strength per assembly simply the product of the unit strength and the active fuel length?

c. Where are the neutron spectrum results for the design basis fuel?

d. Why is the gamma source in Table 3.1-4 (Nuclear and Thermal Parameters) given in terms of MeV/sec instead of photons/sec? It would be useful to have a summary of the totals per cask in Table 3.1-4.

e. With regard to the fission product gas inventory, why include ^{131}Xe ($t_{1/2}$ = 11.92 days) if it is negligible? Why is there no summary of the totals per cask in Table 3.1.5 (Fission Product Gas Inventory)? Are ^{129}I, ^{134}Cs, and ^{137}Cs fission product gas sources?

f. The activation product activities associated with the materials in the head and foot pieces and structural materials are not addressed. Spectral and source information should be provided for these regions as well.

3.2 Structural and Mechanical Safety Criteria

3.2.6 Combined Load Criteria

a. Fifty-six points were selected for stress output, but no statement is made that the highest stress points are guaranteed to be included in these 56 points. It would be easier to assess the stress results if only the highest stresses in each component for each load case, and combination of load cases were reported.

b. On Table 3.2-2 location M_O is described as closure bolt threads. In modeling the entire cask for finite element analysis, how is this level of resolution achieved?

3.3 Safety Protection Systems

3.3.2 Protection by Multiple Confinement Barriers and Systems

3.3.2.1 Acceptance Test Criteria

If no test of the secondary barrier for the upper closure lid can be performed (Table 3.3-2), how can a two barrier system be claimed for the lid? The same applies to the claim of a three barrier system for the penetrations. If no test is performed on the first barrier, how can more than a two barrier system be claimed?

3.3.2.2 Fuel Loading Test Criteria

In Section 3.3.2, a minimum of two barriers to radionuclide migration is specified to provide protection by means of multiple confinement barriers. However, none of the secondary barriers can be pressure tested independently of the primary barrier. This means that a defective secondary seal cannot be detected. A test of the total double-barrier system will only confirm the effectiveness of one seal. Since the secondary seal serves to back up a defective primary seal, we cannot be sure that confinement will be maintained in the event of a primary seal malfunction.

3.3.2.3 Fundamental Leakage Criteria

How is the 10^{-6} atm cm^3/sec leak requirement established? In other words, what A_2 value was used and how was it derived?

3.3.3 Protection by Equipment and Instrumentation Selection

3.3.3.2 Instrumentation

The requirement to discuss "features to provide testability" is not met. How is the cavity pressure transducer system tested for proper operational function?

3.3.4 Nuclear Criticality System

3.3.4.1 Control Methods for Prevention of Criticality

As submitted, prevention of criticality for the storage cask relies on fuel burnup and U-235 depletion. For each initial enrichment of PWR fuel clearly specify the nominal burnup expected, the minimum burnup limit and the fissle materials expected to be

present. Considering that few utilities are currently licensed for handling and storing spent fuel with burnup credit included, provide the following information:

a. A discussion on the benefits, limitations and potential hazards in using credit for burnup. Justify the use of a neutron multiplication factor of .95 limit as having a sufficient safety margin given the additional uncertainties involved with burnup.

b. Specify what administrative controls are required. Does a burnup meter have to be used? Discuss how a utility assures that specific operating histories and minimum burnups of specific fuel bundles and rods are achieved.

c. If burnup credit is required to prevent criticality, discuss how the fuel will be transported on public highways and later disposed. Will the fuel have to be transferred to another cask for transport? This issue should be addressed.

3.3.4.3 Verification Analysis

It is not clear from the description of the analysis whether the verification model assumed homogenized fuel or discrete pins. For large storage casks, a discrete pin model should be used with a 123-group neutron cross-section to properly account for hard neutron spectrum effects.

3.3.5 Radiological Protection

3.3.5.2 Criteria

The information provided in this section is inadequate with regard to the requirements of Regulatory Guide 3.48. Where are the estimates of the annual collective dose for various operations at the ISFSI, e.g. preparation and transfer to storage, inspection, maintenance, etc.? Much of the data presented in Table 5.1-4 (Estimated

Operation Time and Personnel) and Table 7.4-6 (Occupational Doses) seems more appropriate here.

3.3.5.3 Radiological Alarm Systems

Is there an alarm associated with the cavity pressure transducer system? If not, how are personnel alerted in the event of seal failure or a loss of neutron shield?

3.3.6 Fire and Explosion Protection

3.3.6.1 Fire Protection

a. What is the basis for establishing a maximum 10°F temperature increase expected from any site fire?

b. How is the criteria reflected in the analysis for fire?

c. How will the storage cask respond to an 800°C fire for 30 minutes duration?

3.3.6.2 Explosion Protection

The overpressure of 1.0 psi does not seem to be an adequate level for qualifying a cask to resist explosions.

3.3.7 Materials Handling and Storage

3.3.7.1 Spent Fuel Handling and Storage

Handling requirements should address the maximum temperature the fuel casks will attain during the cask drying process after loading.

3.5 Decommissioning Considerations

3.5.1 Storage Casks

What were the input irradiation conditions and materials for the ORIGIN calculation? Was the neutron flux constant throughout the entire wall thickness? Why were the covers and bottom not included? Where is the discussion of the consequences of the cask wall activation? Comparisons of the activation product activities with the limited quantity package limits in 49 CFR 173.435, the limits for Class A waste in 10 CFR 61.55, and the exempt concentrations and exempt quantities in 10 CFR 30.70 and 10 CFR 30.71, respectively, seems appropriate here. Why is the activity at one year after unloading presented? Is one year the maximum time a cask would remain empty before final disposition?

4.0 Installation Design

4.2 Storage Structures

4.2.1.3 Properties of Materials

4.2.1.3.2 Thermal Properties of Materials

a. There is an inconsistency in the density of the solid neutron shield (boro-silicone). From Table 4.2-15 (Thermal Properties of Solid Neutron Shield (Boro-Silicone)) the density is 1.586 gm/cc; from Table 7.3-3 (Shield Material Densities and Compositions) the density is 1.116 gm/cc. Which is correct?

b. The thermal conductivity of both A356 and 5052 aluminum alloy are approximately 60 to 70% lower than that for pure aluminum (Table 4.2-12). It is not conservative to use the thermal conductivity of pure aluminum for heat transfer calculations.

4.2.3 Individual Unit Description

4.2.3.2 Cask Components

4.2.3.2.3 Radiation Protection Components

For the statement "the neutron shields also suppress secondary gamma generation", some brief explanation should be presented.

4.8 Normal Operations

4.8.1 Normal Operation Conditions--Structural Analysis

4.8.1.1 Discussion

No mention of the fuel basket is made. A fuel basket structural analysis must be included for all operating conditions including thermal loads. Demonstrate that the basket at normal operating temperature can sustain the normal operating loads without significant deformations. Since the creep properties of aluminum are significant between 200^{o} and $400^{o}F$, the long term behavior of the fuel basket should be assessed for its combined thermal and loading environment.

4.8.1.2 Weights and Centers of Gravity

"All of the values tabulated in Table 4.8-1 are rounded upward to ensure that all of the analyses which depend upon the cask weight are conservative." We would prefer actual weights since in some cases higher weights could be unconservative.

4.8.1.3 Cask Body Analysis

4.8.1.3.1 Discussion - Loads - Methods of Analysis

Only two bounding thermal cases were listed. The minimum ambient environment (-40°C) coupled with maximum heat decay load (new fuel rods) may not be covered by the two cases listed.

4.8.1.3.2 Finite Element Model Description

a. The weight of the model shown on page 8.2-69 is 126,985 lbs., compared to 160,169 lbs. listed on page 4.8-5. Was the same computer model used for all axisymmetric geometries? If so, was the wrong density input, or the wrong geometry, in order to model the weight more than 20% less than it should be?

b. More detail would be helpful in the description of the finite element model of the separate components. For example, the bolts are said to be modeled with beam elements. These elements should be identified in Figure 4.8-8. Also, describe in detail how the interface is handled between these elements and the closure lid, and the cask body. Also explain how the axisymmetric model deals with the bolt loads.

c. Explain how the "gap elements" describe the interface between the lead and the inner and outer shells.

4.8.1.3.3 Detailed Analysis

a. The equations for P_{rtr} and P_{cr} appear to be incorrect. Furthermore, the distance between supports should be 155.30 rather than 159.8.

b. Note typographical error in title of third paragraph on page 4.8-32.

4.8.1.3.4 Inner Shell Stability Analysis

On page 4.8-61, T/R should be L/R.

4.8.1.5 Closure Valve and Instrumentation Feedthrough Cover Analysis

4.8.1.5.1 Internal Pressure

Clarify as to whether the increase in internal helium gas pressure is 32.4 psig or 32.4 psia? Is the helium backfill pressure atmosphere absolute or gauge?

4.8.1.6.1 Internal Pressure Analysis

a. Include a stress analysis for the shield tank wall while it is in a horizontal position.

b. What is the significance of the factor of 1.5 in the last equation on page 4.8-88?

c. The equation for S_e does not appear to be correct.

d. S_s on page 4.8-91 should be S_e.

4.8.1.6.2 Internal Vacuum Analysis

a. Margin of safety equation is not defined.

b. Units should be in lbs., not in/lb. on page 4.8-93.

4.8.1.7 Storage Support Skid Analysis

a. Symbols like A_t, A_s, A_{by} etc., need to be defined. This practice should be followed throughout the report.

b. The figure on page 4.8-111 is unclear. All figures should be captioned.

4.8.2.2 Thermal Properties of Materials

The equation for the coefficient in the vertical storage mode is different in the third edition of Ref. 4.9.36. Discuss the validity of these equations and the uncertainties associated with them.

4.8.2.3 Thermal Analysis Models

4.8.2.3.1 Cask Body

All dimensions and materials used in the thermal analysis should be shown in Fig. 4.8-15.

4.8.2.3.2 Fuel Storage Basket

a. The aluminum storage basket model needs to show all dimensions (particularly gaps) and the equivalent thickness of homogeneous material.

b. How were the fuel bundles modeled as heat sources?

4.8.2.3.3 Maximum Fuel Rod and Temperature

a. What parameters were used in the fuel rod analysis? Identify all parameters and their values used in the WOOTEN-EPSTEIN correlation.

b. What is the maximum pressure in the fuel rods due to rod pre-pressurization and fission gas release? This data should be included in Table 3.1-1.

c. Provide a plot of temperature vs. time in storage for the hottest fuel rod. This information is needed to estimate the extent of cladding degradation.

d. Show the input and output for heating 5 analysis and the thermal model.

4.8.2.6 Normal Operations Conditions - Multi-Cask Arrays

Provide a description and dimension on the MULTICASK program especially with regard to what modeling assumptions were made and whether or not the program has been benchmarked.

4.8.3 Normal Operations Conditions - Criticality Analysis

4.8.3.2 Keno Model

The use of a .95 neutron multiplication factor limit requires an improved model that includes:

a. Actual cask and basket configurations and dimensions with appropriate clearances. A square cask model may not be conservative.

b. A discrete pin model with a 123-group neutron cross-section should be used to properly account for hard neutron spectrum effects.

c. Detailed information should be provided on the aluminum and boron sheet thicknesses. Although they may be modeled as a homogeneous sheet, it is preferred to model each sheet separately.

4.8.3.3 Criticality Analysis Results

What computer code was used to calculate the burnup, U235 depletion and the fissle isotope fractions? Show the input and output for the depletion calculation for the 3.7% enrichment example. How many burnup and criticality calculations were performed to provide the enrichment and burnup limits in Table 4.8-37? What are the percentages of the fissle isotopes present? What is the criticality sensitivity to burnup, that is, if a 26,000 MWD/MTU burnup is required but only a 23,000 MWD/MTU is achieved, what is the reactivity?

5.0 Operation Systems

5.1 Operation Description

5.1.1 Narrative Description

5.1.1.1 Initial Receipt

Under operation number six a caution is given concerning breaking the pressure transducer wires. What are the consequences of such a break with respect to repair and acceptance of the pressure transducer system?

5.1.1.2 Fuel Loading

a. How will the burnup of each bundle be verified? Will a burnup meter be used? One hundred percent administrative control is difficult to achieve.

b. How is the temperature of fuel rods controlled during the drying out process?

5.1.1.4 ISFSI Storage

There is a requirement in Reg. Guide 3.48 to "describe the means that will be routinely used during storage to evaluate the condition of the" casks. No discussion is provided. From Table 5.1-4 (Estimated Operation Time and Personnel), it is apparent that annual inspection of each cask is considered. Moreover, the capability for remote monitoring of the internal cask pressure has already been discussed. Why are neither of these discussed here?

5.1.2 Flowsheets

a. We suggest that the simplified flow sheets appearing in Table 5.1-1 (Operations Flowsheet--Initial Cask Receipt), Table 5.1-2 (Operations Flowsheet--Fuel Loading), and Table 5.1-3 (Operations Flowsheet--Preparation and Transfer to ISFSI Storage) be annotated to show what specific operations from Section 5.1.1.1 (Initial Receipt), Section 5.1.1.2 (Fuel Loading), and Section 5.1.1.3 (Preparation and Transfer to ISFSI Storage) are associated with each box.

b. With respect to the times and distances listed in Table 5.1-4 (Estimated Operation Time and Personnel), how were they estimated and are they conservative? For all operations with more than one task, where are the estimates of time, personnel, and distance? These will be necessary when computing the occupational dose.

c. Why is "attach lifting yoke" listed separately from the remainder of the operation number two actions during the "Preparation and Transfer to ISFSI Storage" phase?

5.1.3 Identification of Subjects for Safety Analysis

5.1.3.2 Chemical Safety

NAC states that the neutron shield mixture "requires only normal care in its use". What does this mean? Are routine checks or replacement of the mixture required?

5.1.3.4 Instrumentation

Even though the pressure transducer may not be required for safe operations, its failure does impair the ability of the utility to detect a cask with a leaking seal or one whose neutron shield has been lost.

5.2 Fuel Handling Systems

a. Spent fuel storage operations to include placement, storage surveillance, and removal should be summarized. Reference to and use of materials in Section 5.1.1 (Narrative Description) is appropriate here.

b. Any safety-related features of the spent fuel storage system that provide for safe operation under both normal and off-normal conditions should also be summarized. So too, should any limits that are selected for a commitment to action. Reference to any alarm levels presented in earlier sections is appropriate here.

5.3 Other Operating Systems

5.3.2 Component/Equipment Spares

Where is the NAC response to this section. Any design features associated with repair or maintenance operations performed at the ISFSI site to provide continuity of safety under normal and off-normal conditions should be addressed. Reliability of the pressure transducer system is also of interest here. Design provisions to minimize exposure to radiation during repair operations should also be addressed.

5.4 Operation Support Systems

Some discussion of the instrumentation associated with the pressure transducer system is needed here. Monitoring during normal and off-normal operations and accident conditions and redundancy of safety features to ensure adequate safety should be addressed.

6.0 Waste Confinement and Management

This section requires a summary of the radiological impact of normal operations associated with the gaseous radioactive wastes (effluents). There is a reference to Section 7.4.4 (Airborne Radiation). However, no discussion is presented.

7.0 Radiation Protection

7.1 Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)

7.1.2 Design Considerations

In the discussion under regulatory position 2.i, it is stated that "liquid neutron shield tank leakage will be visually obvious". It is also stated that "no close-proximity inspections of the cask are required". How will the leak be obvious from a distance? Furthermore, how long will the leak be "visually obvious"?

7.1.3 Operational Considerations

In their discussion of off-normal and accident event conditions, NAC states "the worst effect of a man-induced accident would be puncture and loss of the liquid neutron shield". What are the operational procedures in this event?

7.2 Radiation Sources

7.2.1 Characterization of Sources

a. The gamma source strength at 1.495 MeV in Table 7.2-1 (Design Basis Fuel Source Strength) does not agree with the source strength in Table 3.1-3 (Photon Spectrum for Design Basis Fuel). Where is the neutron source spectrum? Where are the gamma source strengths for the head and foot piece regions? Why is the total energy flux per cm of active fuel presented instead of the total photon flux per cm of active fuel?

b. Was any consideration given to subcritical multiplication and the associated fission gammas?

c. It is required that "the sources of radiation...be described in the manner needed as input to the shielding design calculations". Both in this section and Section 3.1.1 (Materials to be Stored), the information presented is inadequate.

7.2.2 Airborne Radioactive Material Sources

a. What are the references for the various release fractions? Why are ^{129}I, ^{134}Cs, and ^{137}Cs not considered?

b. What about the requirement to describe the "provisions made for personnel protective measures"? Are any provisions made? If not, why not?

7.3 Radiation Protection Design Features

7.3.1 Installation Design Features

Since the storage is AR and there are no structures, many requirements are negated. However, specific activity, physical and chemical characteristics, and expected concentrations of all sources in Section 7.2 are to be provided. Other requirements appear to be the design radiation dose rate for the storage area and the maintenance and repair activity, and estimates of the radioactive materials that might be discharged during storage. Reference to specific sections of the report where the information may be found would be acceptable. The requirement to indicate how the guidance of (or alternatives to) regulatory position 2 of Regulatory Guide 8.8 will be followed is not fulfilled.

7.3.2 Shielding

What flux-to-dose conversion factors were used in the shielding calculations? Were those of ANSI/ANS-6.1.1 employed? Are the rectangular parallelepiped source regions used in the QAD-CG 3D calculations homogenized? With regard to the gamma dose buildup factors that were used, more information needs to be provided. Why are the product of the material buildup factors not used instead of water for the side and iron for the remaining regions? The reasons for all assumptions should be substantiated.

7.3.2.1 Analysis Source Description

a. Should not the reaction on ^{241}Am be (a,n) instead of (q, n)?

b. NAC states that "eighty-five to ninety-five percent of the neutron source is from spontaneous fissions of ^{242}Cm and ^{244}Cm." Table 7.2-1 (Design Basis Fuel Source Strength) shows 86 percent for $^{242,244}Cm$ and 7 percent for ^{252}Cf. Why the difference? ORIGIN results should be appended.

c. What is the basis for the statement that "the curium spontaneous fission neutron spectrum is quite similar to that of ^{252}Cf". Furthermore, if the ^{252}Cf spontaneous fission spectrum is used to define the neutron source, why is it not presented?

d. Are there differences in sources one, two, three, and four shown in Figure 7.3-2 (QAD-CG Three Dimensional Source Region)? If not, why are they numbered as indicated?

e. Where are the atom densities in atoms/barn-cm for the source region materials in Table 7.3-2 (Source Material Compositions)? Why is the total density for the active fuel source region not presented?

7.3.2.2 Shielding Analysis Dose Points

Actual locations by coordinate position should be presented for the dose point locations shown in Figure 7.3-3 (Detector Locations). Dose points number four and eight are not on the axis and number one is neither in the midplane of the fuel or the shield tank. If the "axial and side points represent the locations for the maximum dose", how were they determined? Would not other dose points have to be used to determine the locations of the maximums?

7.3.2.3 Shielding Analysis Models

a. With respect to Figure 7.3-6 (QAD-CG Three Dimensional Model); what are the coordinates for the fuel region and voids; where is the axial coordinate reference; where are the head piece, foot piece, gas plenum, and active fuel regions of the source region?

b. With respect to Table 7.3-3 (Shield Material Densities and Compositions), where are the atom densities in atoms/barn-cm for the shield materials, where is the C in the ethylene glycol and the K in the potassium tetraborate for the shield tank, where is the Si in the silicon rubber, why is the cumulative density for the silicon rubber inconsistent with the density in Table 4.2-15 (Thermal Properties of Solid Neutron Shield (Boro-Silicone), and why is the cumulative density for the stainless steel (7.306 g/cc) different from the density of most stainless steels of approximately 7.9 g/cc?

7.3.2.3 Shielding Analysis Results - Surface Dose Rates

Are the dose rates really for a point at the indicated location or an average about the location? What are the uncertainties associated with the dose rate results? (Shouldn't this section be numbered 7.3.2.4?)

7.4 Estimated On-Site Collective Dose Assessment

7.4.1 Analysis Methodology

How was the 20 meter cutoff between 1/r and $1/r^2$ determined? At the breakpoint, the difference at the side is 4.3 mrem/hr at 19.9 meter (65.3 feet) versus 0.24 mrem/hr at 20.1 meters (69.2 feet)--a factor of 18. We believe that a 1/r approximation for the dose at the side will be conservative (over-estimated) between 5 and 20 meters. We also have reason to believe that the dose will not be conservative (underestimated) using a $1/r^2$ approximation between 20 and 200 meters. Thereafter, the dose may be conservative (over-estimated) by as much as a factor of 100 at 1000 meters. The approach for calculating the off-surface dose should be validated.

7.4.2 Analysis Results

We disagree with the NAC use of the term "ISFSI boundary" for the 100 meter location. "Controlled area boundary" is a more accurate term. It also will eliminate some confusion when discussing the dose at the site boundary. In Table 7.4-2 (Doses-Single Cask), Table 7.4-3 (Doses-4 Cask Square Array), Table 7.4-4 (Doses-10 Cask Linear Array), and Table 7.4-5 (Doses-180 Cask Maximum Array), there is confusion because the term "ISFSI boundary" is used for the 100 meter location and the point where the annual dose meets the requirements of 10 CFR 72.67(a) (25 mrem/yr). With respect to the boundary location for the 25 mrem/yr requirement, NAC should show how the location was computed for the multiple cask arrays.

7.4.3 Occupational Dose

The personnel exposure time history reference, Table 5.4-4, should be Table 5.1-4 (Estimated Operation Time and Personnel). With respect to Table 7.4-6 (Occupational Doses); how are the dose rates computed? Is DOSE used? Which dose point location is

used for each task? For multitask operations, what is the duration of exposure at each task and at what distance? There is insufficient information in Table 5.1-4 and Table 7.4-6 to allow verification of the occupational doses in Table 7.4-6. Furthermore, why isn't the occupational dose associated with the optional annual inspection presented here? Are the dose rates in Table 7.4-6 on the order of 10^{-5} really that low?

7.4.4 Airborne Radiation

7.4.4.2 Analysis - Normal Operations Conditions Basis

a. The title for this section is not consistent with the assumption of 10 percent failed fuel rods. From Section 7.2.2 (Airborne Radioactive Material Sources), 10 percent failed fuel rods represents normal and off-normal conditions.

b. The values of A_2 for ^{85}Kr and ^{131}Xe are for an uncompressed gas; by the definition in 10 CFR 71 Appendix A, all gases are compressed. Why are the uncompressed values used?

c. Should the reference to Section 3.2.2.3 on page 7.4-21 be Section 3.3.2.3?

d. How was the downstream pressure for the measured condition set at 2 atmospheres? According to Section 4.8.1.5.1 (Internal Pressure Analysis), "the NAC S/T cask is backfilled with helium at 1 atmosphere".

e. How do A_2 values relate to requirements of 10CFR72. The methods outlined in this section for calculating the A_2 value for a mixture of isotopes is incorrect.

7.4.4.3 Analysis - Off-Normal Operations and Accident Event Conditions

Like the previous section, the title for this section is inconsistent with the assumption of 100 percent failed fuel rods. From Section 7.2.2 (Airborne Radioactive Material Sources), 100 percent failed fuel rods represents accident conditions only. Comments with respect to the values of A_2 and the downstream pressure for the measured condition apply here as well.

7.4.4.4 Boundary Dose

Titles for "Normal" and "Off-Normal/Accident" should be made consistent with earlier definitions for the tables in this section. NAC should give some reason for computing the whole body dose from ^3H inhalation as if it were ^{85}Kr. There is also some question as to whether the diffusion factor is conservative. From Safety Guide 1.25 the diffusion factor is 1.0×10^{-2} at 200 meters and is projected to be on the order of 5×10^{-2} at 100 meters for the same wind speed and Pasquill diffusion category. Why the difference?

7.6 Estimated Off-Site Collective Dose Assessment

Inconsistent reference is again made to a 100 meters boundary. We also believe that some generic estimate of the off-site dose is required.

7.7 NAC S/T Cask Performance

7.7.3 Cask Leakage

NAC states that "Chapter 7.4.4.2 analyzes leakage from the NAC S/T cask for normal and off-normal operations conditions, Chapter 7.4.4.3 for accident conditions". This is consistent with the analyses that appear in the above sections, but not consistent with the titles of the sections themselves.

8.0 Accident Analysis

8.1 Off-Normal Operation

Is this a 10 percent failed fuel rod scenario? According to Section 7.2.2 (Airborne Radioactive Material Sources) it should be.

8.1.1 Event - Leakage Through a Cask Closure

8.1.1.3 Analysis of Effects and Consequences - Leakage Through a Cask Closure

NAC states that "a containment barrier seal does not perform normally (it develops a leak)", but does not specify the magnitude of the leak. They further state that "the internal gas pressure would cause some of the helium gas atmosphere to escape through the closure leak" but give no estimate of the amount that might be lost. Finally they state that "the radiological impact (if any) of the off-normal event is minimal". This statement along with the others must be supported by facts.

8.1.2 Event - Fission Product Gas Release

This cannot be an off-normal event according to Section 7.2.2 (Airborne Radioactive Material Sources); 100 percent failed fuel rods is an accident event.

8.1.2.1.3 Shielding Analysis - Loss of Neutron Shield

a. Comments made earlier in Section 7.3.2.3 (Shielding Analysis Results - Surface Dose Rates) with regard to Table 7.3-4 (Combined Gamma and Neutron Dose Rates) apply to Table 8.2-14 (Combined Gamma and Neutron Dose Rates) as well.

b. We question the analysis procedure for computing a distance where the requirements of 10 CFR 72.68(b) are met for the various cask array scenarios. Since minimum distances were established earlier in Section 7.4.2 (Analysis Results) to meet the 10 CFR 72.67(a) requirement, why were not the doses computed at these distances for the various arrays? This seems a much more logical procedure.

8.1.2.3 Analysis of Effects and Consequences - Fission Product Gas Release

a. A summary of the results cited from Section 7.4.4 (Airborne Radiation) should be presented to substantiate the statement that "leakage and airborne radiation are far below established criteria".

b. How was the total free gas per rod value on page 8.1-8 determined?

8.1.2.4 Corrective Action - Fission Product Gas Release

Since there will be some loss of gaseous fission products with leakage, the NAC statement that the "cask safely contains all fission products released from the fuel rods" is incorrect.

8.1.3 Event - Failure of Instrumentation

8.1.3.4 Corrective Actions - Failure of Instrumentation

a. We do not agree with the NAC position that no corrective action is required in the event the pressure transducer system fails.

b. Can an event that ruptures the neutron shield lead to storage conditions that are outside the operating limits? If so, then a failed pressure transducer system should be repaired or replaced.

8.1.4 Radiological Impact from an Off-Normal Event

a. The statement that "fission product gas release" will "have no radiological impact" because it does not "involve leakage from the cask" is not correct. There is an established leak rate. The extent of the leakage is also dependent on the amount of time it takes to move the cask from the storage to fuel handling area.

b. The use of the 100 meter "ISFSI boundary" is inconsistent. Why not provide the actual value at 100 meters to substantiate the statement that the radiological impact is "less than the 5 rem criterion".

8.2 Accidents

8.2.1 Accident - Loss of Neutron Shield

8.2.1.1 Cause of Accident - Loss of Neutron Shield

How will a leak be detected if the casks are monitored remotely? A close-proximity visual inspection would seem to be required.

8.2.1.2 Accident Analysis - Loss of Neutron Shield

Will there be a pressure increase as a result of loss of the neutron shield? If so, it would seem that this is a direct structural consequence of neutron shield loss. The decrease in allowable stress limits and increased thermal stresses would also seem to be direct structural consequences of the neutron shield loss.

8.2.1.2.1 Thermal Analysis - Loss of Neutron Shield

a. Table 4.8-14 is not the correct reference for loss of neutron shield thermal analysis. It should be table 4.8-17.

b. How is the heat transfer across the neutron shield annulus calculated?

8.2.1.2.3 Shielding Analysis - Loss of Neutron Shield

a. What are repair procedures in the event the neutron shield is lost?

b. What are the occupational dose consequences of neutron shield loss?

c. Provide thermal analysis for event of neutron shield loss.

8.2.1.3 Accident - Cask Burial Under Debris

8.2.1.3.2 Accident Analysis - Cask Burial Under Debris

Provide a thermal model for the cask burial analyses.

8.2.2 Accident-Fire

Is there any possibility that fire could cause the neutron shield to rupture or the lead to slump? If so, then radiological consequences must be addressed.

8.2.3 Accident - Cast Tip over

8.2.3.1 Cause of Accident - Cask Tip over

The statement, "There are no credible means by which the NAC S/T cask could experience a tip over" is not true, as discussed in Section 8.2.10 below.

8.2.3.2 Accident Analysis - Cask Tip over

a. Is there any possibility that cask tip over could cause some rupture of the fuel rods or lead slump? If so, then radiological consequences must be addressed.

b. A statement is made that the tip over such that the C.G. moves through 58.71 vertical inches is not as severe as a 55 g oblique drop accident. This statement requires a great deal of supporting discussion since it is not obvious.

8.2.3.2.1 Structural Analysis - Tipping Against Another Cask

a. The statement is made that if a cask tips and strikes a horizontal cask, the most severe angle is 45°. The corresponding figure, Fig. 8.2-2, is unclear.

b. The notation used on page 8.2-55 is confusing. Is there a difference between m and M? What about r and R?

c. The calculation of moment of inertia is incorrect:

$$I = (0.25\ mr^2 + 0.33\ mR^2) + MR^2$$

is given, this is the formula for a solid cylinder. The cask is more closely a hollow cylinder.

d. The rotational kinematics discussion in this section is not correct.

8.2.3.3 NAC S/T Cask Performance - Cask Tip over Accident

The statement is made that if the neutron shield tank is lost and the fuel ruptures entirely, the ISFSI site boundary dose is less than the criteria of 10 CFR 72. Be more specific. What is the dose at how many meters for a single tipped cask and for an array of casks?

8.2.4 Accident - Cask Drop

8.2.4.1 Discussion

The statement "a design impact force of 55g on the cask is established" should define what is meant by 55g. A discussion of an impact limiter and a maximum drop height satisfying this requirement would be appropriate.

8.2.4.2 Detailed Analysis

a. Is there any possibility that a cask drop could cause some rupture of the fuel rods or lead slump? If so, then radiological consequences must be addressed.

b. Is the loading obtained by applying a body force equivalent to 55 times the unit mass in the direction of impact? In other words, is this a static calculation? Our work at LLNL indicates that inertially applied impact loadings may be nonconservative.

c. The statement is made that the cask contents is not included in the analysis because it produces a negligible stress of 37 psi at the contact surface. We calculate, roughly, 745 psi due to the cask contents at 55 g's.

8.2.4.2.2 Side Drop

a. Is there any supporting evidence (experimental or theoretical) which indicates that the load distribution shown in Figure 8.2-4 is correct? In order to evaluate the approach used, it is necessary to answer this question. In addition, a statement should be made that sufficient terms have been used in the fourier expression. This could list, for example, the maximum stresses and strains calculated for each component for each harmonic.

b. A sample of the plots described on page 8.2-146, used to adjust the radial stresses, should be included.

8.2.4.2.3 Analysis - Oblique Drop

The statement is made that "slapdown" is less severe than initial impact due to the larger impact area. In fact, large impact areas generally imply higher stresses. The possibility that the "slapdown" as a secondary impact may increase the stresses, should be addressed.

8.2.5 Accident - Flood

8.2.5.2 Accident Analysis - Flood

8.2.5.2.2 Structural Analysis

a. Would not the tip over of a vertically stored cask or roll-over of a horizontally stored cask cause the neutron shield tank to collapse?

b. The notation on page 8.2-266 is confusing. Please define ρ_V, P_V, and $* P_V$. The reference to page 3-55 of the Mark's Mechanical Engineering Handbook should be deleted, and a reference to Bernoulli's equation included instead. Also, list the assumption used in the application of Bernoulli's equation.

c. If the buoyancy effect of the water is taken into account, the overturning moment would be lower and, as a consequence, the water velocity for overturning.

8.2.6 Accident - Cask Seal Leakage

8.2.6.3 NAC S/T Cask Performance - Cask Seal Leakage

a. As discussed in Section 7.4.4.4 (Boundary Dose), NAC should state why the whole body dose from ^{3}H inhalation was computed as ^{85}Kr.

b. Comments about the diffusion factor are also relevant here. If the more conservative value of 5.0×10^{-2} were used, the dose from a single cask would increase from 192.3 mrem to 794.6 mrem.

8.2.7 Accident - Tornado Winds

See comment regarding Bernoulli's equation in Section 8.2.5.2.2.

8.2.8 Accident - Tornado Missiles

8.2.8.1 Cause of Accident - Tornado Missiles

It seems inconceivable that the impact of massive missile on the cylindrical cask can result in a uniform external pressure.

8.2.8.2 Accident Analysis - Tornado Missiles

8.2.8.2.1 Analysis - Massive Missile Vertical Cask - Tip over

a. On page 8.2-297 there is a reference to Fig. 8.2-1. This seems to be the wrong figure number.

b. On this same page, $W_c h$ is called the final potential energy. This is the initial potential energy.

c. The angular velocity symbol is missing on page 8.2-294.

d. Isn't W_i the angular velocity of the cask after impact?

e. It would appear that the impact force on the cask is a function of deceleration of the missile rather than the force necessary to maintain equilibrium.

8.2.8.2.3 Analysis - Protective Barrier Missile Vertical Cask Tip-Over

The paragraph appearing at the top of page 8.2-361 is identical to the final paragraph on page 8.2-357. Since, the latter appears in Section 8.2.8.2.2 (Analysis - Penetrant Missile Vertical Cask Tip-Over) we wonder about the accuracy of the former.

8.2.10 Accident - Earthquakes

8.2.10.2 Accident Analysis - Earthquake Natural Frequency Analyses

This section is unnecessary.

Tip over Analysis - Cask Vertical

Five cases are considered with various combinations of horizontal and vertical accelerations. Only one case is relevant, called Case 5 here. In this case, however, the tipping moment is incorrectly calculated, taking credit for the stabilizing weight of the cask twice. When this is corrected the conclusion is reversed. The cask will tip in the event of the cited earthquake.

Roll-over Analysis - Cask Horizontal

Again, the only relevant case is case 5, which is incorrectly calculated.

Again, the conclusion is reversed. The cask will roll off of its skid during a D.B.E. east of the Rockies under the worst conditions.

8.4 References - Chapter 8

Some references are missing. They are probably on page 8.4-1 which is also missing.

10.0 Operating Controls and Limits

10.1 Proposed Operating Controls and Limits

10.1.2 Bases for Operating Controls and Limits

The topical report assumes that dynamic loading on the cask will result in a "g" load not exceeding 55 g. Consequently, operation limits should be established such that the cask will not experience a deceleration that results in a "g" load exceeding 55 g.

10.1.2.1 Fuel Characteristic Limit

a. Why is the gamma flux in Table 10.1-1 (Operating Controls and Limits) in MeV/sec instead of photons/sec? The distance from the cask to boundary of 100 meters is incorrect. A minimum distance of 187.2 meters is required for a single cask. Why is this distance not presented here? We also question whether the atmospheric diffusion factor is conservative.

b. On page 10.1-4, 1×10^6 should be 1×10^{-6}.