72-1027 TN-68 Amendment 1
TABLE OF CONTENTS

SECTION **PAGE**

TABLE OF CONTENTS

SECTION **PAGE**

CHAPTER 11

ACCIDENT ANALYSES

This Chapter describes the postulated off-normal and accident events which could occur during storage of the TN-68 cask at an ISFSI. Detailed analysis of the events are provided in other SAR chapters and referenced herein.

11.1 Off-Normal Operations

Off-normal operations are design events of the second type (Design Event II) as defined in ANSI/ANS 57.9[(1)]. Design Event II conditions consist of that set of events that, although not occurring regularly, can be expected to occur with moderate frequency or on the order of once during a calendar year of ISFSI operation.

Two off-normal conditions have been considered with regard to the TN-68 cask, a loss of electric power and cask seal leakage.

11.1.1 Loss of Electric Power

A total loss of electric power to the ISFSI is postulated. The failure could be either an open or a short to ground circuit, or any other mechanism capable of producing an interruption of power.

11.1.1.1 Postulated Cause of the Event

A loss of power to the ISFSI may occur as a result of natural phenomena, such as lightning or extreme wind, or as a result of undefined disturbances in the nonsafety-related portion of the electric power system.

If electric power is lost, the following systems could be de-energized and rendered nonfunctional:

- Area lighting
- Cask pressure monitoring instrumentation

11.1.1.2 Detection of Event

A loss of power at the ISFSI site would be detected during periodic surveillance by noting that area lighting is not operational.

11.1.1.3 Analysis of Effects and Consequences

This event has no safety or radiological consequences because a loss of power will not affect the integrity of the storage casks, jeopardize the safe storage of the fuel, or result in radiological releases. None of the systems whose failure could be caused by this event are necessary for the accomplishment of the safety function of the cask. The lighting system functions merely for convenience and visual monitoring, and the instrumentation monitors the long-term performance of the storage casks with respect to the cask seals. None of these parameters are expected to change rapidly and their status is not dependent upon electric power.

A loss of power has no effect on the subcritical condition of the cask, cask confinement or retrievability of the fuel.

11.1.1.4 Corrective Actions

Following a loss of electric power to the ISFSI, plant maintenance personnel will be informed and will isolate the fault and restore service by conventional means. Such an operation is straightforward and routine for the maintenance personnel of an electric utility.

11.1.1.5 Radiological Impact from Off-Normal Operations

No radiological impact from off-normal operations is postulated.

11.1.2 Cask Seal Leakage or Leakage of the Overpressure Monitoring System

The storage casks feature redundant seals in conjunction with an extremely rugged body design. Additional barriers to the release of radioactivity are presented by the sintered fuel pellet matrix and the zircaloy cladding which surrounds the fuel pellets. Furthermore, the interseal gaps are pressurized in excess of the cask cavity. As a result, no credible mechanisms that could result in leakage of radioactive products have been identified. However, to bound the worst off-normal event, leakage of one seal is evaluated.

11.1.2.1 Postulated Cause of the Event

A combined event of failure of one of the seals in addition to a failure of the pressure monitoring system is assessed. This could also be a failure of the pressure boundary of the overpressure system in addition to a failure of the monitoring alarm system.

11.1.2.2 Detection of Event

Detection of a seal leak in addition to the loss of the pressure monitoring system would be by means of periodic testing or maintenance.

11.1.2.3 Analysis of Effects and Consequences

Analysis has been performed in Chapter 3 to show that the lid bolts will prevent leakage of the seals during normal and postulated accident events. A description of the three possible leaks which could occur is presented below:

- In any of the inner confinement seals (lid seal, inner vent seal or inner drain seal).

  The lid and lid penetration cover bolts and seals are designed to prevent leakage during all normal, off-normal and postulated accident events. Therefore this is a very unlikely event.

  In this case the overpressure system, which has a higher pressure than the cask cavity, would leak helium into the cask cavity. Since the pressure is higher in the overpressure tank, it would prevent leakage of radioactive materials out of the cask cavity until the pressure between the overpressure tank and the cask cavity equalized. This would take several years, depending on the size of the leak. At the test leak rate, the overpressure system pressure would always exceed the cask cavity pressure, as shown in Chapter 7. Therefore no leakage of radioactive material can occur, even if the alarm were to fail. Chapter 7 also demonstrates that even if the inner seal has experienced a latent seal failure there is ample time for identifying the leak through routine surveillances.

- In any of the outer seals (lid, overpressure port cover, vent cover or drain cover)

  The lid and lid penetration cover bolts and seals are designed to prevent leakage during all normal, off-normal and postulated accident events. Therefore this is a very unlikely event.

  In this case, leakage out of the interspace to the atmosphere would occur. This would not result in release of radioactive material from the cask cavity since the inner seal is intact. Again, as demonstrated in Chapter 7, a latent seal failure of the outer seals would not result in a release of any radioactive material to the environment. There is also ample time for identifying the leak through routine surveillances.

- A leak in the overpressure system

  This is the most likely cause of a leak, since it is a non safety related component and not designed to withstand accident loadings.

  In this case two scenarios could exist:

  - The overpressure system is not functioning and the inner seal is intact. In this case there is no release of radioactive material to the environment; or

  - The overpressure system is not functioning and the inner seal is leaking at some rate.

In this latter case, leakage out of the interspace to the atmosphere and the cask cavity would occur. This would not result in release of radioactive material from the cask cavity until the pressure fell to the cask cavity pressure. At the test leak rate of 1 x 10^{-5} ref cm^3/s, this would not occur during the 20 year storage period.

However, a leak of this magnitude in combination with a loss of the over pressure system has been evaluated as both an off-normal and accident condition in Section 7.3.

The results of these calculations assuming off-normal conditions indicate that an individual located at the site boundary (100 m from the cask) for an entire year would receive an effective dose equivalent of 5.39 mrem, a thyroid dose of 1.07 mrem, a bone surface dose of 17.8 mrem, and a lung dose of 22.5 mrem. These doses are below the regulatory limits of 10 CFR 72.104(a) of 2.5 x 10^{-1} mSv (25 mrem) to the whole body, 7.5 x 10^{-1} mSv (75 mrem) to the thyroid and 2.5 x 10^{-1} mSv (25 mrem) to any other critical organ.

The results of these calculations assuming accident conditions indicated that at the site boundary (100m from the cask), for a 30 day release, the total effective dose equivalent is 175 mrem. The total organ dose equivalent to any individual organ (the critical organ in this case is the bone surface) is 927 mrem for a 30 day release. The lens dose equivalent to the lens of the eye is 176 mrem for a 30 day release. These values are well below the limiting off site doses defined in 10 CFR 72.106(b).

Another accident condition under consideration is that the overpressure system is not functioning and the inner seal has experienced a latent seal failure. This analysis is presented in 7.3.3. This accident analysis demonstrates that a latent failure up to 100 times greater that the test value could occur and there is ample time for recovery before the limiting off site doses in 10 CFR 72.106(b) are met. The probability that a gross leak of an inner seal in combination with a gross leak in the outer seal is not considered a credible event.

11.1.2.4 Corrective Action

The overpressure system leak would be repaired at the ISFSI depending on the complexity of the repair, or the cask would be returned to the spent fuel pool for seal replacement. Repairs which could be performed at the ISFSI are tightening of the fittings, replacement of valves or switches, localized weld repairs or replacement of components.

11.1.2.5 Radiological Impact

For the worst case, which includes loss of alarm, complete loss of the pressure differential between the cask and the overpressure system, and complete loss of the overpressure system pressure boundary, the dose rates at the site boundary would increase as stated above, but are below the regulatory limits of 10 CFR 72.104(a).

11.1.3 Overpressure Tank Needs Refilling

The overpressure tank may need to be refilled during the cask storage period to ensure that a positive pressure differential is maintained between the overpressure system and the cask cavity. This maintenance will be performed by plant personnel as scheduled maintenance.

11.1.3.1 Postulated Cause of the Event

Slow leakage of the outer or inner seals, less than the allowable leak rate.

11.1.3.2 Detection of Event

Pressure monitoring system alarm would indicate that the pressure in the overpressure tank had fallen below the set point. The set point is generally set much higher than the maximum cavity pressure so that there is sufficient time to repressurize the tank. Calculations performed in Chapter 7 (See Figure 7.1-1) show that it would take 11 years to reach the alarm setpoint if the cask were leaking at the seal test leakage acceptance rate of 1×10^{-5} ref cm^3/s. Plant maintenance procedures will be developed to ensure that the tank pressure will be verified or repressurized at least once per ten years.

11.1.3.3 Analysis of Effects and Consequences

The overpressure tank may need to be repressurized during the storage period. This event has no safety or radiological consequences because the set point of the pressure monitoring system is selected so that there is ample time to repressurize prior to any leakage out of the cask cavity.

11.1.3.4 Corrective Actions

After repressurizing, the overpressure tank will be checked to ensure no leakage around the fittings. The pressure transducers/switches will be checked for operability.

11.1.3.5 Radiological Impact

Estimated operational doses due to this action is included in Chapter 10.

11.2 Accidents

Accidents are design events of the third and fourth type (Design Events III and IV) as defined in ANSI/ANS 57.9. Design Event III consists of that set of infrequent events that could reasonably be expected to occur during the lifetime of the ISFSI. Design Event IV consists of the events that are postulated because their consequences may result in the maximum potential impact on the immediate environs. Their consideration establishes a conservative design basis for certain systems with important confinement features.

11.2.1 Earthquake

11.2.1.1 Cause of Accident

The design earthquake (DE) is postulated to occur as a design basis extreme natural phenomenon. The cask is evaluated for a safe shutdown earthquake (SSE) of 0.26g horizontal and 0.17g vertical.

11.2.1.2 Accident Analysis

Cask response to a seismic event is evaluated in Section 2.2.3 and Appendix 3A. Results of these analyses show that the cask does not tip over or slide and that the containment vessel stresses resulting from the seismic loads are below ASME code allowable stresses for accident conditions. The leak-tight integrity of the cask is not compromised. No damage to the cask is postulated. The basket stresses are also low and do not result in deformations that would prevent fuel from being unloaded from the cask.

11.2.1.3 Accident Dose Calculations

The DE does not damage the cask. Hence, no radioactivity is released and there is no associated dose increase due to this event.

11.2.1.4 Corrective Actions

After a seismic event, the cask would be inspected for damage. Any debris would be removed. An evaluation would be performed to determine if the cask were still within the licensed design basis. The pressure monitoring system would be tested, and repaired if necessary. If necessary, the cask would be returned to the spent fuel pool for unloading.

11.2.2 Extreme Wind and Tornado Missiles

11.2.2.1 Cause of Accident

The extreme winds due to passage of the design tornado as defined in Section 2.2.1 are postulated to occur as an extreme natural phenomenon.

11.2.2.2 Accident Analysis

In section 2.2.1, it is shown that extreme winds do not result in a cask tip over or sliding of the cask. The pressure due to high winds on the surface of the cask is bounded by the assumed external pressure of 25 psi. The stresses in the cask resulting from this external pressure are presented in Appendix 3A. High winds have no effect on the leak tight integrity of the cask, and do not result in damage to the cask. High winds do not affect the basket or the ability to retrieve the spent fuel from the cask. The effect of tornado missiles hitting the cask has been evaluated in Section 2.2.1. These analyses show that the stresses in the cask as a result of missile impact are well below the ASME Code allowable stresses for Accident (Level D) conditions. It is also shown in Section 2.2.1 that the tornado missile impact will not result in cask tipover. Local damage to the neutron shield may result from the tornado missile impact. The cask may slide about 7.3 inches due to missile impact below the CG of the cask, however, the space between the two casks is more than 90 inches, therefore, the cask will not impact each other. Table 5.1-2 provides the surface dose rates of the cask assuming that the neutron shield is completely removed. This data can be used by the site to conservatively determine the maximum dose rates at the site boundary due to tornado missile impact.

11.2.2.3 Accident Dose Calculations

Extreme winds are not capable of overturning the casks nor of damaging the cask seals. The overpressure system and the neutron shielding may be damaged. To determine the bounding dose, loss of neutron shielding (Section 11.2.5.3) is combined with the total effective dose equivalent (TEDE) from the loss of one confinement barrier and 100% fuel cladding failure (Section 11.2.9.3). The resulting site boundary accident dose, 888 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b).

11.2.2.4 Corrective Actions

After excessive high winds or a tornado, the cask would be inspected for damage. Any debris would be removed. Any damage resulting from impact with a missile would be evaluated to determine if the cask were still within the licensed design basis. The pressure monitoring system would be tested, and repaired if necessary. If necessary, the cask would be returned to the spent fuel pool for unloading.

11.2.3 Flood

11.2.3.1 Cause of Accident

Natural event.

11.2.3.2 Accident Analysis

The postulated floods and high water levels are discussed in Section 2.2.2. The analysis presented shows that the cask will withstand the external pressure due to the flood and the velocity of the flowing water will not result in a cask tip or cause the cask to slide.

11.2.3.3 Accident Dose Calculations

The probable maximum flood is not capable of overturning the casks or of damaging their seals. The overpressure system and the neutron shielding may be damaged. To determine the bounding dose, loss of neutron shielding (Section 11.2.5.3, 713 mrem) is combined with the total effective dose equivalent (TEDE, 175 mrem) from the loss of one confinement barrier and 100% fuel cladding failure (Section 11.2.9.3). The resulting site boundary accident dose, 888 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b).

11.2.3.4 Corrective Actions

After a flood of the ISFSI site, the casks would be inspected for damage. The surfaces of the cask would potentially need to be cleaned and repainted in local areas. Any debris would be removed. If there were any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis.

11.2.4 Explosion

11.2.4.1 Cause of Accident

Explosion in the general vicinity.

11.2.4.2 Accident Analysis

Regulatory Guide 1.91provides guidance for Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants. This document states that an acceptable method for demonstrating that a nuclear power plant has the ability to withstand the possible effects of explosions occurring on transportation routes is to demonstrate that the rate of exposure to a peak positive incident overpressure in excess of 1 psi is less than 10^{-6} per year. Further a review of several utility ISFSI Safety Analysis Reports indicate that explosions of a few psi or less are postulated as the worst case explosion.

The TN-68 cask is a robust steel design, and an explosion would not be expected to damage the cask. For conservatism, the cask is evaluated for an external pressure of 25 psi. The cask body is a thick walled construction and is capable of withstanding very high external loads without collapse.

11.2.4.3. Accident Dose Calculations

The cask will not tip as a result of the postulated pressure wave. Accordingly, no cask damage or release of radioactivity is postulated. Since no radioactivity is released, no resultant dose increase is associated with this event.

11.2.4.4 Corrective Actions

After an explosion in the vicinity of the ISFSI site, the casks would be inspected for damage. The surfaces of the cask would potentially need to be cleaned and repainted in local areas. Any debris would be removed. If there were any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis.

11.2.5 Fire

11.2.5.1 Cause of Accident

Combustible materials will not normally be stored at an ISFSI. Therefore, a credible fire would be very small and of short duration such as that due to a fire or explosion from a vehicle or portable crane.

However a hypothetical fire accident is evaluated for the TN-68 cask based on a fuel fire, the source of fuel being that from a ruptured fuel tank of the cask transporter tow vehicle. The bounding capacity of the fuel tank is 200 gallons and the bounding hypothetical fire is an engulfing fire around the cask.

11.2.5.2 Accident Analysis

The evaluation of the hypothetical fire event is presented in Section 4.4 of the SAR. The fire thermal evaluation is performed primarily to demonstrate the containment integrity of the TN-68. This is assured as long as the metallic lid seals remain below 536°F and the cavity pressure is less than 100 psig.

Based on the thermal analyses for the fire accident conditions, the TN-68 packaging can withstand the hypothetical fire accident event without compromising its containment integrity. No melting of the metallic cask components occurs. Peak cask component temperatures are summarized in Table 4.4-1. The maximum seal temperature is calculated to be 470°F which is well below the temperature limit of the metallic seals. The average cavity gas temperature peaks at 572°F and the pressure increases to 71.7 psig (5.88 atm abs). See Section 4.7.5. The pressure inside the cask cavity is well below the design pressure of 100 psig.

The neutron shield will off-gas during the hypothetical accident. A pressure relief valve is provided on the outer shell to prevent the pressurization of the outer shell. Shielding analyses have been performed showing acceptable consequences even if all the resin disappears. (See Chapter 5).

11.2.5.3 Accident Dose Calculations

Local damage to the neutron shielding may result from the fire. This is bounded by removal of all the neutron shielding which is evaluated in Chapter 5. Even with this conservative assumption, the site boundary accident dose rates are below 5 rem to the whole body or any organ as specified in 10CFR72.106(b).

The off-site doses are evaluated for two accident conditions:

1) loss of radial neutron shielding
2) loss of the protective cover and top neutron shield.

For accident conditions, the following assumptions are made:

a) the nearest postulated site boundary is 100 meters distant from the cask
b) the accident involves a single cask
c) the accident duration is 30 days
d) a person remains at the postulated site boundary 24 hours per day for the entire duration

The normal condition total dose rates at 100 meters are scaled by the ratio of accident to normal surface dose rates as shown in the following table. All units are mrem/hr.

	normal dose rate 100 m Table 5.1-3	accident, surface Table 5.1-2	normal, surface Table 5.1-2	accident, 100 m mrem/hr
gamma	4.0562E-02	774	98.0	3.204E-01
neutron	6.7362E-03	2032	20.6	6.677E-01
			Total:	9.881E-01

The total dose over 30 days would be 711 mrem. The background from the rest of the ISFSI would be 1/12 of the 25 mrem/year limit (10 CFR 72.104), or 2 mrem. The combined total accident dose would be 713 mrem.

11.2.5.4 Corrective Actions

After a fire, the cask would be inspected for damage. The surfaces of the cask would potentially need to be cleaned and repainted in local areas. The neutron shielding material may have been damaged during the fire. If there is any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis. If the cask is no longer within the design basis, the cask will be returned to the spent fuel pool and unloaded. The neutron shield may need to be replaced prior to putting the cask back into service.

11.2.6 <u>Inadvertent Loading of a Newly Discharged Fuel Assembly</u>

11.2.6.1 <u>Cause of Accident</u>

The possibility of a spent fuel assembly, with a heat generation rate greater than 0.441 kW, being erroneously selected for storage in a cask has been considered. The cause of this accident is postulated to be an error during the loading operations, e.g., wrong assembly picked by the fuel handling crane, or a failure in the administrative controls governing the fuel handling operations.

11.2.6.2 <u>Accident Analysis</u>

The fuel assemblies require several years of storage in the spent fuel pool before the heat generation decays to a rate below 0.441 kW. In addition, the shielding analysis assumes that the fuel has been cooled at least 7 years prior to loading. This accident scenario postulates the inadvertent loading of an assembly not intended for storage in the cask, with a heat generation rate in excess of the design basis specified in Section 2.1.

In order to preclude this accident from going undetected, and to ensure that appropriate corrective actions can take place prior to the sealing of the casks, a final verification of the assemblies loaded into the casks and a comparison with fuel management records is required to assure that the correct assemblies are loaded.

These administrative controls and the records associated with them will be included in the procedures described in Chapter 8.

Appropriate and sufficient actions will be taken to ensure that an erroneously loaded fuel assembly does not remain undetected. In particular, the storage of a fuel assembly with a heat generation in excess of 0.441 kW is not considered credible in view of the multiple administrative controls. Also, surface radiation dose measurements will provide final verification that a newly discharged fuel assembly has not been loaded in the cask.

There is no thermal or shielding analysis impact since the improperly loaded cask will not get out of the water due to independent review. Criticality is not a concern provided that the initial enrichment limit is not exceeded. The loading of a higher enriched fuel assembly is evaluated as a separate accident in section 11.2.7.

11.2.6.3 <u>Accident Dose Calculations</u>

The inadvertent loading of a newly discharged fuel assembly not intended for storage is prevented by administrative control. Therefore, no resultant radiation dose increases would occur.

11.2.6.4 Corrective Actions

If it is determined that a fuel assembly has been loaded which is outside the bounds of the design basis, it shall be removed from the cask.

11.2.7 Inadvertent Loading of a Fuel Assembly with a higher initial enrichment than the Design Basis Fuel

11.2.7.1 Cause of Accident

The possibility of a spent fuel assembly with initial enrichment greater than permitted by the Technical Specifications has been considered. The cause of this accident is postulated to be an error during the loading operations, e.g., wrong assembly picked by the fuel handling crane, or a failure in the administrative controls governing the fuel handling operations.

11.2.7.2 Accident Analysis

This accident is prevented by administrative controls specified in the operations in Chapter 8. Prior to loading of the fuel, the basket type must be verified by the cask serial number, and the pre-selected fuel must be checked to verify that each bundle's enrichment is at or below the limit specified for that basket type.

In order to preclude this accident from going undetected, and to ensure that appropriate corrective actions can take place prior to the sealing of the casks, a final verification of the assemblies loaded into the casks and a comparison with fuel management records is required to assure that the correct assemblies are loaded.

These administrative controls and the records associated with them will be included in the procedures described in Chapter 8.

Appropriate and sufficient actions will be taken to ensure that an erroneously loaded fuel assembly does not remain undetected.

11.2.7.3 Accident Dose Calculations

The inadvertent loading of a fuel assembly with higher initial enrichment than the design basis is prevented by administrative control.

The criticality safety evaluations provide a large margin of conservatism by the assumption that the fuel is unirradiated and contains no burnable poison. The 0.95 limit on k_{eff} for normal, off-normal, and credible accident conditions provides further safety margin. An evaluation of loading a 5% enriched 10x10 fuel assembly at the center of the basket designed for 3.7% enrichment shows that k_{eff} increases from 0.9221 to 0.9260, a change of 0.004, less than 10% of the 0.05 safety margin (Table 6.4-3). Therefore, in the event that a fuel assembly with higher initial enrichment were loaded, the fuel would remain well below critical.

There is no resultant dose rate increase due to this condition.

11.2.7.4 Corrective Actions

If it is determined that a fuel assembly has been loaded which is outside the bounds of the design basis, it shall be removed from the cask.

11.2.8 Hypothetical Cask Drop and Tipping Accidents

11.2.8.1 Cause of Accident

The stability of the TN-68 storage cask in the upright position on the ISFSI concrete storage pad is demonstrated in Section 2.2 of this SAR. The effects of tornado wind and missiles, flood water and earthquakes are described in Sections 2.2.1, 2.2.2 and 2.2.3, respectively. It is shown in those sections that the cask will not tip over under the most severe natural phenomena specified in this Topical Safety Analysis Report.

The cask is designed for single failure proof lifting at the reactor site.

An 18 inch vertical cask drop is postulated to occur during handling while the cask is moved onto or off of a transport vehicle. The trunnions are designed to the requirements of ANSI N14.6[(2)] for lifting devices. The cask will generally be handled by a transport vehicle in a vertical orientation and not lifted higher than 18 in. Other drop events which may be postulated at a specific ISFSI site will be evaluated in accordance with 10 CFR 72.212.

This section of the SAR considers design events of the third and fourth types (includes accidents) as defined in ANSI/ANS 57.9. The third type of events are those that could reasonably be expected to occur over the lifetime of the ISFSI (does not include tipping of the cask). The fourth type of events include severe natural phenomena (described in Section 11.2.1 through 11.2.5) and man induced low probability events postulated because their consequences could result in the maximum potential impact on the immediate environs. Therefore the cask is examined for both dropping and tipping accidents, which are hypothetical impact events that are extremely unlikely to occur.

11.2.8.2 Accident Analyses

The cask is evaluated under bottom end impact on the ISFSI storage pad after a drop from a height of 18 inches in Section 3A.2.3.2. The storage pad is the hardest concrete surface outside of the spent fuel storage building. The cask is generally oriented vertically and not lifted higher than 18 in. once it leaves the containment building. Therefore this case is an upper bound drop event since impact onto a softer surface would result in lower cask deceleration and a lower impact force. The cask is also evaluated under a tipover event on the storage pad even though (as demonstrated in Section 2.2) the cask can not tip over. Appendix 3D determines the maximum g loading which would result for a cask end drop and a tipover. The maximum

deceleration due to an 18 inch bottom end drop is 55.5 g's. The maximum deceleration due to a tipover accident is 65 g's.

The cask is analyzed conservatively for an 60 g vertical load simulating the end drop, and a 65g side drop conservatively simulating the tipover. The analyses are presented in Section 3A.2.3.2. The cask stresses for the cask tip over and drop event are reported in Tables 3A.2.5-15 through -26. All stresses meet the design criteria. An additional analysis of a cask tipping over and impacting on the trunnions is evaluated in 3A.2.4.3. This analysis shows that the local stresses around the trunnion are acceptable, and the g loadings are less severe than the side drop analyzed in 3A.2.3.2.

The stresses in the lid bolts due to the two postulated drop accidents are presented in Section 3A.3.2. This analysis shows that the stresses in the bolts due to the accident loads are well below the allowable limit of $3S_m$ and the bolt yield strength.

The stresses in the basket due to the two postulated drop accidents are presented in Appendix 3B. These analyses show that the basket is structurally satisfactory under the tipover and end drop loads.

Depending on site constraints and requirements, there may be handling conditions different than those analyzed above. For example, there may be a need to lift the cask higher than 18 inches over an impact limiter or a surface which is softer than the ISFSI concrete pad. Prior to using the cask at these sites, 10 CFR 72.212 evaluations shall be performed to ensure that the g loading on the cask is bounded by the g loadings presented above.

11.2.8.3 Accident Dose Calculations

Cask tip will not breach the cask confinement barrier. No radioactivity will be released and no resultant doses will occur.

To determine the bounding dose, the loss of neutron shielding (Section 11.2.5.3) is combined with the total effective dose equivalent (TEDE) from the loss of one confinement barrier and 100% fuel cladding failure (Section 11.2.9.3). The resulting site boundary accident dose, 888 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b).

This conservatively bounds any damage to the neutron shield as a result of a cask tip over or drop event.

11.2.8.4 Corrective Actions

After a tipover or cask handling drop, the cask would be inspected for damage. The neutron shielding material may have been damaged due to impact. If there is any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis. If the cask is no longer within the design basis, the cask will be returned to the spent fuel pool and unloaded. The neutron shield may need to be replaced prior to putting the cask back into service.

11.2.9 Loss of Confinement Barrier

11.2.9.1 Cause of Accident

It is assumed that the overpressure system has stopped functioning and fire conditions exist.

11.2.9.2 Accident Analysis

It is assumed that at least one set of seals is still functioning, and that material can be released at the test leak rate of 1×10^{-5} ref cm^3/s. It is also assumed that all of the fuel rods have failed, and the temperature inside the cask is comparable to the fire accident conditions. The cask is assumed to leak at this rate for 30 days.

In this accident, the confinement function of the fuel rod cladding and one set of seals is eliminated. Heat removal and radiation shielding functions operate in the normal passive manner.

This is equivalent to breaking one cask seal barrier, removing the pressure monitoring system, failing all the cladding in all the loaded fuel assemblies (gap activity release), and finally, failing the fuel pellets themselves. The analysis is presented in Section 7.3.2.

11.2.9.3 Accident Dose Calculations

The dose evaluation due to this postulated accident is given in Section 7.3.2.1. The total effective dose equivalent is 175 mrem. The total organ dose equivalent to any individual organ (the critical organ in this case is the bone surface) is 927 mrem for a 30 day release. The lens dose equivalent to the lens of the eye is 176 mrem for a 30 day release. The shallow dose equivalent to the skin is 1.67 mrem/30 days. These values are well below the limiting off site doses defined in 10 CFR 72.106.

11.2.9.4 Corrective Actions

In the event of cask leakage, the cask would be returned to the spent fuel pool and the seals would be replaced. In addition the overpressure system would be checked to determine the cause of failure and corrective measures to prevent future recurrence would be taken. The overpressure system and pressure monitoring equipment would be repaired or replaced as necessary prior to returning the loaded cask to the ISFSI for storage.

11.2.10 Buried Cask

11.2.10.1 Cause of Accident

Earthquake or other natural phenomenon resulting in collapse of building, other structure or other manmade or earthen material onto a cask.

11.2.10.2 Accident Analysis

An evaluation was made to determine the increase in cask temperature with time assuming the cask was completely buried in a medium which will not provide the equivalent cooling of natural convection and unrestricted radiation to the environment. The details of this analysis are provided in Section 4.4.

The results of this analysis show that if the cask is not uncovered within 0.6 hours, the neutron shield temperature will exceed the allowable long term temperature limit of 300°F (149°C). The cavity pressure, including the contribution due to 100 % fuel failure, will exceed 100 psig at approximately 76 hours. The cask seal temperature will not reach its 536°F (280°C) limit at this time. The fuel temperature off-normal limit of 1058°F (570°C) is reached about 90 hours after burial occurs.

11.2.10.3 Accident Dose Calculations

Slow degradation of the neutron shielding would begin to occur shortly after burial resulting in higher surface dose rates. At about 76 hours, the cask internal pressure exceeds the design pressure of 100 psig. The seals will not reach their long term maximum temperature of 536°F (280°C) even 120 hours after burial occurs (see Table 4.4-2). In the event that the cask could not be unburied after 76 hours, release of radioactive gases could occur.

To determine the bounding dose, loss of neutron shielding (Section 11.2.5.3) is combined with the total effective dose equivalent (TEDE) from the loss of one confinement barrier and 100% fuel cladding failure (Section 11.2.9.3). The resulting site boundary accident dose, 888 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b).

11.2.10.4 Corrective Actions

The cask should be unburied as soon as possible to prevent release of radioactive material. The cask will be inspected for damage. The neutron shielding material may have been damaged during the burial. If there is any damage, an evaluation would be performed to determine if the cask were still within the licensed design basis. If the cask is no longer within the design basis, the cask will be returned to the spent fuel pool and unloaded. The neutron shield and all seals would need to be replaced prior to putting the cask back into service.

11.3 REFERENCES

1. American Nuclear Society, ANSI/ANS-57.9, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), 1992.

2. American National Standards Institute, ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 pounds or More, 1986.