TN-68 GENERIC TECHNICAL SPECIFICATIONS

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1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CHANNEL OPERATIONAL TEST (COT)	A CHANNEL OPERATIONAL TEST (COT) shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the operability of required alarm functions. The COT shall include adjustments, as necessary, of the alarm setpoint so that the setpoint is within the required range and accuracy.
INTACT FUEL ASSEMBLY	An INTACT FUEL ASSEMBLY is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Partial fuel assemblies, that is spent fuel assemblies from which fuel rods are missing, shall not be classified as INTACT FUEL ASSEMBLIES unless dummy fuel rods are used to displace an amount of water equal to or greater than that displaced by the original fuel rod(s).
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a cask while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the cask and end when the cask is supported from the transporter.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the Independent Spent Fuel Storage Installation (ISFSI) while a cask containing spent fuel is sitting on a storage pad within the ISFSI.

(continued)

1.1 Definitions (continued)

TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on a cask loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the cask is first suspended from the transporter and end when the cask is at its destination and no longer supported from the transporter.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include al licensed activities on a cask while fuel assemblies are being unloaded. UNLOADING OPERATIONS begin when the cask is no longer supported by the transporter and end when the last fuel assembly is removed from the cask.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.
EXAMPLES	The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify... AND A.2 Restore...	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop... OR A.2.1 Verify... AND A.2.2.1 Reduce... OR A.2.2.2 Perform... OR A.3 Remove...	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the cask. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the cask is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the cask is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1	Perform Action B.1.	12 hours
	AND		
	B.2	Perform Action B.2.	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND to complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One system not within limit.	A.1	Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1	Perform Action B.1.	12 hours
	AND		
	B.2	Perform Action B.2.	36 hours

When a system is determined to not meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

-------------------------NOTE-------------------------
Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. LCO not met.	A.1	Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Perform Action B.1.	12 hours
	AND		
	B.2	Perform Action B.2.	36 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

(continued)

1.3 Completion Times (continued)

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed", constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria. SR 3.0.4 restrictions would not apply if both the following conditions are satisfied:

a. The Surveillance is not required to be performed; and

b. The Surveillance is not required to be met or, even if required to be met, is not known to be failed.

(continued)

1.4 Frequency (continued)

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications(TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the cask is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the cask is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

If the interval as specified by SR 3.0.2 is exceeded while the cask is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2.

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

2.0 FUNCTIONAL AND OPERATIONAL LIMITS

2.1 Functional and Operational Limits

2.1.1 Fuel to be Stored in the TN-68 Cask

The spent nuclear fuel to be stored in the TN-68 cask shall meet the following requirements:

A. Fuel shall be unconsolidated INTACT FUEL ASSEMBLIES.

B. Fuel shall be limited to fuel with Zircaloy cladding.

C. Fuel shall be limited to the following fuel types with the following unirradiated specifications:

Assembly Type		Designation	#of Fuel Rods	Max Rod Pitch	Min Rod OD	Max Uranium Content (MTU/assy)
GE	7x7	2,2A,2B	49	0.738	0.563	0.1977
GE	7x7	3,3A,3B	49	0.738	0.563	0.1896
GE	8x8	4,4A,4B	63	0.640	0.493	0.1880
GE	8x8	5,6,6B,7,7B	62	0.640	0.483	0.1876
GE	8x8	8,8B	62	0.640	0.483	0.1885
GE	8x8	8,8B,9,9B,10	60	0.640	0.463	0.1824
GE	9x9	11,13	74	0.566	0.440	0.1757
GE	10x10	12	92	0.510	0.404	0.1857

Fuel designs 6, 6B, 7 and 7B may also be designated as P, B or BP. Fuel designs may be C, D or S lattice only.

D. Fuel assemblies may be channeled or unchanneled. Channel thickness up to 0.120 inches thick are acceptable.

E. Fuel assemblies shall have the bounding characteristics as specified in Table 2.1.1-1 and below:

i. 3.7 wt% U-235 maximum initial lattice-average enrichment.
ii. The maximum heat load per assembly shall not exceed 0.312 kW.
iii. The maximum weight per individual assembly shall be 705 pounds.

(continued)

2.0 FUNCTIONAL AND OPERATIONAL LIMITS (continued)

2.2 Functional and Operational Limits Violations

If any Functional and Operational Limit of 2.1.1 is violated, the following actions shall be completed:

2.2.1 The affected fuel assemblies shall be removed from the cask and placed in a safe condition.

2.2.2 Within 24 hours, notify the NRC Operations Center.

2.2.3 Within 30 days, a special report shall be submitted to the NRC which describes the cause of the violation and the actions taken to restore compliance and prevent recurrence.

Table 2.1.1-1

Minimum Acceptable Cooling Time as a
Function of Max. Burnup and Min. Initial Enrichment

REQUIRED BWR COOLING TIMES (YEARS)

Min.Init. Enrich. (1)	Maximum Burnup (GWd/MTU) (2)										
(bundle ave %w)(3)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	10	10										
1.1	10	10										
1.2	10	10										
1.3	10	10										
1.4	10	10										
1.5	10	10	10	10	11	11	11					
1.6	10	10	10	10	10	11	11	11				
1.7	10	10	10	10	10	11	11	11	12			
1.8	10	10	10	10	10	11	11	11	11	12		
1.9	10	10	10	10	10	11	11	11	11	12		
2.0	10	10	10	10	10	10	11	11	11	12	12	
2.1	10	10	10	10	10	10	11	11	11	12	12	12
2.2	10	10	10	10	10	10	11	11	11	12	12	12
2.3	10	10	10	10	10	10	11	11	11	11	12	12
2.4	10	10	10	10	10	10	10	11	11	11	12	12
2.5	10	10	10	10	10	10	10	11	11	11	12	12
2.6	10	10	10	10	10	10	10	11	11	11	12	12
2.7	10	10	10	10	10	10	10	10	11	11	11	12
2.8	10	10	10	10	10	10	10	10	10	11	11	12
2.9	10	10	10	10	10	10	10	10	10	11	11	12
3.0	10	10	10	10	10	10	10	10	10	10	11	12
3.1	10	10	10	10	10	10	10	10	10	10	11	12
3.2	10	10	10	10	10	10	10	10	10	10	10	11
3.3	10	10	10	10	10	10	10	10	10	10	10	10
3.4	10	10	10	10	10	10	10	10	10	10	10	10
3.5	10	10	10	10	10	10	10	10	10	10	10	10
3.6	10	10	10	10	10	10	10	10	10	10	10	10
3.7	10	10	10	10	10	10	10	10	10	10	10	10

■ - not evaluated

1. Round actual value down to next lower tenth.
2. Round actual value up to next higher GWd/MTU.
3. Average over entire bundle including uranium blankets.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
LCO 3.0.3	Not applicable to a cask.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of the cask. Exceptions to this are stated in the individual Specifications. These exceptions allow entry into specified conditions in the Applicability when the associated ACTIONS to be entered allow operation in the specified condition in the Applicability only for a limited period of time.
LCO 3.0.5	Equipment removed from service or declared to not meet the LCO to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate that the LCO is met.
LCO 3.0.6	Not applicable to a cask.
LCO 3.0.7	Not applicable to a cask.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once", the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, And the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of a cask.

3.1 CASK INTEGRITY

3.1.1 Cask Cavity Vacuum Drying

LCO 3.1.1 The cask cavity vacuum drying pressure shall be sustained at or below 4 mbar absolute for a period of at least 30 minutes after isolation from the vacuum drying system.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-------NOTE------- Not applicable until SR 3.1.1.1 is performed. ---------------------- A. Cask cavity vacuum drying pressure limit not met.	-------NOTE----------- Action A.1 applies until helium is removed for subsequent operations. ------------------------- A.1 Achieve or maintain a nominal helium environment in the cask. <u>AND</u> A.2 Establish cask cavity drying pressure within limits.	12 hours 96 hours
B. Required Action A.1 and associated Completion Time not met.	B.1 Remove all fuel assemblies from the cask.	7 days
C. Required Action A.2 and associated Completion Time not met.	C.1 Remove all fuel assemblies from the cask.	30 days

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.1.1	Verify that the equilibrium cask cavity vacuum drying pressure is brought to ≤ 4 mbar absolute for ≥ 30 minutes	Once, within 36 hours of completion of cask draining.

3.1 CASK INTEGRITY

3.1.2 Cask Helium Backfill Pressure

LCO 3.1.2 The cask cavity shall be filled with helium to a pressure of 2.0 atm absolute (+0/-10%).

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-------NOTE------- Not applicable until SR 3.1.2.1 is performed. ------------------ A. Cask initial helium backfill pressure limit not met.	-------NOTE----------- Action A.1 applies until helium is removed for subsequent operations ---------------------- A.1 Achieve or maintain a nominal helium environment in the cask <u>AND</u> A.2 Establish cask cavity backfill pressure within limits.	 6 hours 48 hours
B. Required Action A.1 and Associated Completion Time not met.	B.1 Remove all fuel assemblies from the cask.	7 days
C. Required Action A.2 and associated Completion Time not met.	C.1 Remove all fuel assemblies from the cask.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify that the cask cavity helium pressure is 2.0 atm absolute (+0/-10%).	Once, within 42 hours of completion of cask draining.

3.1 CASK INTEGRITY

3.1.3 Cask Helium Leak Rate

LCO 3.1.3 The combined helium leak rate for all closure seals shall not exceed 1.0 E-5 ref-cc/sec.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
------NOTE------- Not applicable until SR 3.1.3.1 is performed. --------------------- A. Cask helium leak rate not met.	A.1 Establish cask helium leak rate within limit.	48 hours
B. Required Action A.1 and Associated Completion Time not met.	B.1 Remove all fuel assemblies from cask.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify cask helium leak rate is within limit.	Once, prior to TRANSPORT OPERATIONS.

3.1 CASK INTEGRITY

3.1.4 Combined Helium Leak Rate

LCO 3.1.4 The combined helium leak rate for all closure seals and the overpressure system shall not exceed 1.0 E-5 ref-cc/sec.

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

-----------------------NOTE-----------------------
Separate Condition entry is allowed for each cask.

CONDITION	REQUIRED ACTION	COMPLETION TIME
-------NOTE------- Not applicable until SR 3.1.4.1 is performed. ------------------ A. Combined helium leak rate not met.	A.1 Establish combined helium leak rate within limit.	48 hours
B. Required Action A.1 and Associated Completion Time not met.	B.1 Remove all fuel assemblies from cask.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----------------NOTE----------------- This surveillance may be combined with SR 3.1.3.1. ----------------------------------	Once, prior to TRANSPORT OPERATIONS
SR 3.1.4.1 Verify the combined helium leak Rate is within the limit.	<u>OR</u>
	Once within 48 hours of commencing STORAGE OPERATIONS.

3.1 CASK INTEGRITY

3.1.5 Cask Interseal Pressure

LCO 3.1.5 Cask interseal pressure shall be maintained at a pressure of at least 3.0 atm absolute.

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

---------------------------NOTE---------------------------

Separate Condition entry is allowed for each cask.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Cask interseal pressure below limit.	A.1 Restore cask interseal pressure above limit.	7 days
B. Required Action A.1 and Associated Completion Time not met.	B.1 Remove all fuel assemblies from cask.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify cask interseal helium pressure above limit.	7 days
SR 3.1.5.2	Perform a CHANNEL OPERATIONAL TEST (COT) to verify proper functioning of pressure switch/transducer on cask overpressure system.	Once, within 7 days of commencing STORAGE OPERATIONS AND 36 months thereafter

3.1 CASK INTEGRITY

3.1.6 Cask Minimum Lifting Temperature

LCO 3.1.6 The loaded cask shall not be lifted if the outer surface of the cask is below -20°F.

APPLICABILITY: During TRANSPORT OPERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Cask surface temperature below limit.	A.1 Lower cask to safe position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-- NOTE: This surveillance does not need to be performed if temperature is known to be above freezing. -- SR 3.1.6.1 Verify outer surface temperature is above limit.	Once, immediately prior to lifting cask and prior to cask transfer to or from ISFSI

3.2 CASK RADIATION PROTECTION

3.2.1 Cask Surface Contamination

LCO 3.2.1 Removable contamination on the cask exterior surfaces shall not exceed:

a. 1000 dpm/100 cm^2 (0.2 Bq/cm^2) from beta and gamma sources; and
b. 20 dpm/100 cm^2 (0.003 Bq/cm^2) from alpha sources.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
------NOTE------- Not applicable until SR 3.2.1.1 is performed. --------------------- A. Removable contamination on the cask exterior surface exceeds either limit.	A.1 Decontaminate cask surfaces to below required levels.	Prior to TRANSPORT OPERATIONS.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify that the removable contamination on the exterior surface of the cask does not exceed the specified limits.	Once, prior to TRANSPORT OPERATIONS

4.0 DESIGN FEATURES

The Specifications in this section include the design characteristics of special importance to each of the physical barriers and to maintenance of safety margins in the cask design. The principle objective of this category is to describe the design envelope that constrains any physical changes to essential equipment. Included in this category are the site environmental parameters which provide the bases for design, but are not inherently suited for description as LCOs.

4.1 Storage Cask

4.1.1 Criticality

The design of the storage cask, including spatial constraints on adjacent assemblies (minimum basket opening of 5.97 inches by 5.97 inches) and boron content of the basket material (minimum areal density equal to 0.030 g B10/cm^2 for borated aluminum or 0.036 g B10/cm^2 for B_4C/aluminum composite) shall ensure that fuel assemblies are maintained in a subcritical condition with a k_{eff} of less than 0.95 under all conditions of operation.

4.1.2 Structural Performance

The cask has been evaluated for a cask tipover (equivalent to a side drop of 65 g's) and a bottom end drop resulting in an axial gravitational (g) loading of 60 g's.

4.1.3 Codes and Standards

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, 1995 Edition with Addenda through 1996, is the governing Code for the TN-68 Cask. The TN-68 cask confinement boundary is designed, fabricated and inspected in accordance with Subsection NB of the ASME Code to the maximum practical extent. Exceptions to the code are listed in Table 4.1-1.

The TN-68 basket is designed, fabricated and inspected in accordance with Subsection NG of the ASME Code to the maximum practical extent. Exceptions to the code are listed in Table 4.1-1.

Proposed alternatives to ASME Code Section III, 1995 Edition with Addenda through 1996 including exceptions allowed by Table 4.1-1 may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or Designee. The applicant should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or

2. Compliance with the specified requirements of ASME Code Section III, 1995 Edition with Addenda through 1996, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for exceptions in accordance with this section should be submitted in accordance with 10 CFR 72.4.

4.1.4 Helium Purity

The cask shall be filled with helium with a purity of at least 99.99%. This level of purity will ensure that the residual impurities in the cask cavity will be less than 1 mole.

4.2 Storage Pad

4.2.1 Storage Locations for Casks

Casks shall be spaced a minimum of 16 feet apart, center to center. This minimum spacing will ensure the proper dissipation of radiant heat energy from an array of casks as assumed in the TN-68 Safety Analysis Report.

4.3 ISFSI Specific Parameters and Analyses

ISFSI specific parameters and analyses that shall need verification by the system user are, as a minimum, as follows:

1. Tornado maximum wind speeds: 360 mph

2. Flood levels up to 57 feet and drag forces up to 45,290 lbs.

3. Seismic loads on the ISFSI pad of up to 0.26g horizontal and 0.17g vertical.

4. Average daily ambient temperatures: ≥ -20°F minimum; ≤ 100°F maximum

5. The potential for fires and explosions shall be addressed, based on site-specific considerations. Fires and explosions should be bounded by the cask design bases parameters of 200 gallons of fuel (in the tank of the transporter vehicle) and an external pressure of 25 psig.

6. Supplemental Shielding: In cases where engineered features (i.e. berms, shield walls) are used to ensure that the requirements of 10 CFR 72.104(a) are met, such features are to be considered Important to Safety and must be evaluated to determine the applicable Quality Assurance Category.

Table 4.1-1
TN-68 ASME Code Exceptions

The cask confinement boundary is designed, fabricated and inspected in accordance with the ASME Code Subsection NB to the maximum practical extent. The basket is designed, fabricated and inspected in accordance with ASME Code Subsection NG to the maximum practical extent. The gamma shielding, which is primarily for shielding, but also provides structural support to the confinement boundary during accident events, was designed in accordance with Subsection NF of the code. Inspections of the gamma shielding are performed in accordance with ASME code Subsection NF as detailed in the SAR. Other cask components, such as the protective cover, outer shell and neutron shielding are not governed by the ASME Code.

Component	Reference ASME Code/Section	Code Requirement	Exception, Justification & Compensatory Measures
TN-68 Cask	NB-1100/ Subsection NCA NB-2000	Stamping and preparation of reports by the Certificate Holder, Surveillances, Use of ASME Certificate Holders	The TN-68 cask is not N stamped, nor is there a code design specification or stress report generated. A design criteria document is generated in accordance with TN's QA Program and the design and analysis is performed under TN's QA Program and presented in the SAR. The cask may also be fabricated by other than N-stamp holders and materials may be supplied by other than ASME Certificate holders. Surveillances are performed by TN and utility personnel rather than by an Authorized Nuclear Inspector (ANI)
TN-68 Cask	NCA-3800	QA Requirements	The quality assurance requirements of NQA-1 or 10 CFR 72 Subpart G are imposed in lieu of NCA-3800 requirements.
Lid Bolts	NB-3232.3	Fatigue analysis of bolts	A fatigue analysis of the bolts is not performed for storage, since the bolts are not subject to significant cyclical loads.

Table 4.1-1
TN-68 ASME Code Exceptions

Component	Reference ASME Code/Section	Code Requirement	Exception, Justification & Compensatory Measures
Confinement Vessel	NB-6200	Hydrostatic Testing	The confinement vessel is hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200 with the exception that the confinement vessel is installed in the gamma shield shell during testing. The confinement vessel is supported by the gamma shield during all design and accident events.
Weld of bottom inner plate to the confinement shell	NB-5231	Full penetration corner welded joints require the fusion zone and the parent metal beneath the attachment surface to be UT after welding.	The required UT inspection will be performed on a best efforts basis. The joint will be examined by RT and either PT or MT methods in accordance with ASME Subsection NB requirements. The joint may be welded after the confinement shell is shrink fitted into the gamma shield shell. The geometry of the joint may not allow for UT inspection.
Confinement Shell Rolling Qualification	NB-4213	The rolling process used to form the inner vessel should be qualified to determine that the required impact properties of NB-2300 are met after straining by taking test specimens from three different heats.	If the plates are made from less than three heats, each heat will be tested to verify the impact properties.

Table 4.1-1
TN-68 ASME Code Exceptions

Component	Reference ASME Code/Section	Code Requirement	Exception, Justification & Compensatory Measures
Confinement Vessel	NB-7000	Vessels are required to have overpressure protection	No overpressure protection is provided. Function of confinement vessel is to contain radioactive contents under normal, off-normal and accident conditions of storage. Confinement vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
Confinement Vessel	NB-8000	Requirements for nameplates, stamping and reports per NCA-8000	TN-68 cask is to be marked and identified in accordance with 10 CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Transnuclear approved QA program.
Confinement Vessel	NB-1131	The design specification shall define the boundary of a component to which other component is attached.	A code design specification was not prepared for the TN-68 cask. A TN design criteria was prepared in accordance with TN's QA program. The confinement boundary is specified in Chapter 1 of the SAR.
Basket poison and aluminum plates	NG-2000	Use of ASME Materials	The poison material and the aluminum plates are not used for structural analysis, but to provide criticality control and heat transfer. They are not code materials.

Table 4.1-1
TN-68 ASME Code Exceptions

Component	Reference ASME Code/Section	Code Requirement	Exception, Justification & Compensatory Measures
Basket Rails	NG-2000	Use of ASME Materials	The fuel basket rail material is not a Class 1 material. It was selected for its properties. Aluminum has excellent thermal conductivity and a high strength to weight ratio. NUREG-3854 and 1617 allow materials other than ASME Code materials to be used in the cask fabrication. ASME Code does provide the material properties for the aluminum alloy up to 400°F and also allows the material to be used for Section III applications (Class 2 and 3). The construction of the aluminum rails will meet the requirements of Section III, Subsection NG.
Basket Compartment longitudinal weld joint (3/16" thick)	NG-5231	Table NG-3352-1 specifies that in order to utilize a quality factor of 0.9 for a full penetration weld, examination must be in accordance with NG-5231. For this 3/16" thick weld, NG-5231 specifies that either a liquid penetrant or magnetic particle examination be performed "of the root, each subsequent layer, and on the external weld surfaces and adjacent base material for ½" on each side of the weld."	The purpose of imposing a progressive examination is to ensure that the weld deposit in successive layers is sound. The weld joint in question is a one pass, full penetration, automatic machine weld without using any filler material, (PAW). Therefore, the weld is only a thin, one layer weld. Liquid penetrant examination is performed in accordance with NG-5111 and NG-5231 on the single pass weld. This allows a quality factor of 0.9 to be utilized.

5.0 ADMINISTRATIVE CONTROLS

5.1 Training Module

Training modules shall be developed under the general licensee's training program as required by 10 CFR 72.212(b)(6). Training modules shall require a comprehensive program for the operation and maintenance of the TN-68 spent fuel storage cask and the independent spent fuel storage installation (ISFSI). The training modules shall include the following elements, at a minimum:

- TN-68 cask design (overview)
- ISFSI Facility design (overview)
- Systems, Structures, and Components Important to Safety (overview)
- TN-68 Dry Storage Cask Safety Analysis Report (overview)
- NRC Safety Evaluation Report (overview)
- Certificate of Compliance conditions
- TN-68 Technical Specifications
- Applicable Regulatory Requirements (e.g.,10 CFR72, Subpart K, 10CFR 20, 10 CFR Part 73)
- Required Instrumentation and Use
- Operating Experience Reviews
- TN-68 Cask Operating and Maintenance procedures, including:
  Fuel qualification and loading
  Rigging and handling
  Loading Operations as described in Chapter 8 of the SAR
  Unloading Operations including reflooding as described in Chapter 8 of the SAR
  Auxiliary equipment operations and maintenance (i.e. vacuum drying, helium backfilling and leak testing, reflooding)
  Transfer operations including loading and unloading of the Transport Vehicle
  ISFSI Surveillance operations
  Radiation Protection
  Maintenance
  Security
  Off-normal and accident conditions, responses and corrective actions.

5.0 ADMINISTRATIVE CONTROLS

5.2 Programs

The following programs shall be established, implemented, and maintained.

5.2.1 Cask Sliding Evaluation

The TN-68 cask has been evaluated for sliding in the unlikely events of storm winds, missile impacts, flood forces and earthquakes. A static coefficient of 0.35 is used in these analyses. This program provides a means for evaluating the coefficient of friction to ensure that the cask will not slide during the seismic event.

a. Pursuant to 10 CFR 72.212, this program shall evaluate the site-specific ISFSI pad configurations/conditions to ensure that the cask would not slide during the postulated design basis earthquake. The program shall conclude that the surface static friction coefficient of friction is greater than or equal to 0.35.

b. Alternatively, for site-specific ISFSI pad configurations/conditions with a lower coefficient of friction than 0.35, the program shall evaluate the site specific conditions to ensure that the TN-68 cask will not slide during the postulated design basis earthquake. The program shall also evaluate storm winds, missile impacts and flood forces to ensure that the cask will not slide such that it could result in impact with other casks or structures at the ISFSI. The program shall ensure that these alternative analyses are documented and controlled.

5.2.2 Cask Transport Evaluation Program

This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met.

a. Pursuant to 10 CFR 72.212, this program shall evaluate the site-specific transport conditions. The program shall evaluate the site-specific conditions to ensure that the end-drop loading does not exceed 60g. The program shall ensure that these analyses are documented and controlled.

b. This program shall establish administrative controls and procedures to ensure that cask TRANSPORT OPERATIONS are conducted within the limits imposed by the Technical Specifications or the alternative analysis described above.

5.2.3 Cask Surface Dose Rate Evaluation Program

This program provides a means to help ensure that ISFSI's using TN-68 casks do not violate the requirements of 10 CFR Part 72 and Part 20 regarding radiation doses and dose rates. The TN-68 design incorporates the use of an optional shield ring above the radial neutron shield. This shield ring may be installed to ensure that the dose rates meet the requirements identified below. The shield ring does not need to be installed on casks which have been surveyed and meet the dose rate limits identified in 5.2.3.2 and 5.2.3.3 without installation of the shield ring.

1. As part of its evaluation pursuant to 10 CFR 72.212, the licensee shall perform an analysis to confirm that the limits of 10 CFR Part 20 and 10 CFR 72.104 will be satisfied under the actual site conditions and configurations considering the planned number of casks to be used and the planned fuel loading conditions.

2. On the basis of the analysis in TS 5.2.3.1, the licensee shall establish a set of cask surface dose rate limits which are to be applied to TN-68 casks used at the site. Limits shall establish average gamma-ray and neutron dose rates for:

    a. The top of the TN-68 cask (protective cover),
    b. The sides of the radial neutron shield,
    c. The side of the cask above the radial neutron shield, and
    d. The side of the cask below the radial neutron shield.

3. Notwithstanding the limits established in TS 5.2.3.2, the dose rate limits may not exceed the following values as calculated for a content of design basis fuel as follows:

    a. 120 mr/hr gamma and 10 mr/hr neutron on the top (protective cover)
    b. 75 mr/hr gamma and 10 mr/hr neutron on the sides of the radial neutron shield.
    c. 360 mr/hr gamma and 45 mr/hr neutron on the side surfaces of the cask above the radial neutron shield.
    d. 210 mr/hr gamma and 70 mr/hr neutron on the side surfaces of the cask below the radial neutron shield.

4. Prior to transport of a TN-68 containing spent fuel to the ISFSI, the licensee shall measure the cask surface dose rates and calculate average values as described in TS 5.2.3.7 and 5.2.3.8.

    The measured average dose rates shall be compared to the limits established in TS 5.2.3.2 or the limits in TS 5.2.3.3, whichever are lower. When needed to meet this specification (TS 5.2.3), the optional 1-inch thick steel shell above the radial neutron shield is required to be in place.

5. If the measured average surface dose rates do not meet the limits of TS 5.2.3.2 or TS 5.2.3.3, whichever are lower, the licensee shall take the following actions:

   a. Notify the U.S. Nuclear Regulatory Commission (Director of the Office of Nuclear Material Safety and Safeguards) within 30 days.
   b. Administratively verify that the correct fuel was loaded, and
   c. Perform an analysis to determine that placement of the as-loaded cask at the ISFSI will not cause the ISFSI to exceed the radiation exposure limits of 10 CFR Part 20 and 72.

6. If the analysis in 5.2.3.5.c shows that placement of the as-loaded cask at the ISFSI will cause the ISFSI to exceed the radiation exposure limits of 10 CFR Part 20 and 72, the licensee shall remove all fuel assemblies from the cask within 30 days of the time of cask loading.

7. Surface dose rates shall be measured approximately at the following points (see also Figure 5.2.3-1).

   a. Above the Radial Neutron Shield (A): Midway between the top of the cask body flange and the top of the radial neutron shield. At least six measurements equally spaced circumferentially.

   b. Sides of Radial Neutron Shield (B,C,D): one sixth, one half, and five sixths of the distance from the top of the radial neutron shield. At least six measurements equally spaced circumferentially at each elevation, two of which shall be at the circumferential location of the cask trunnions. However, no measurement shall be taken directly over the trunnion.

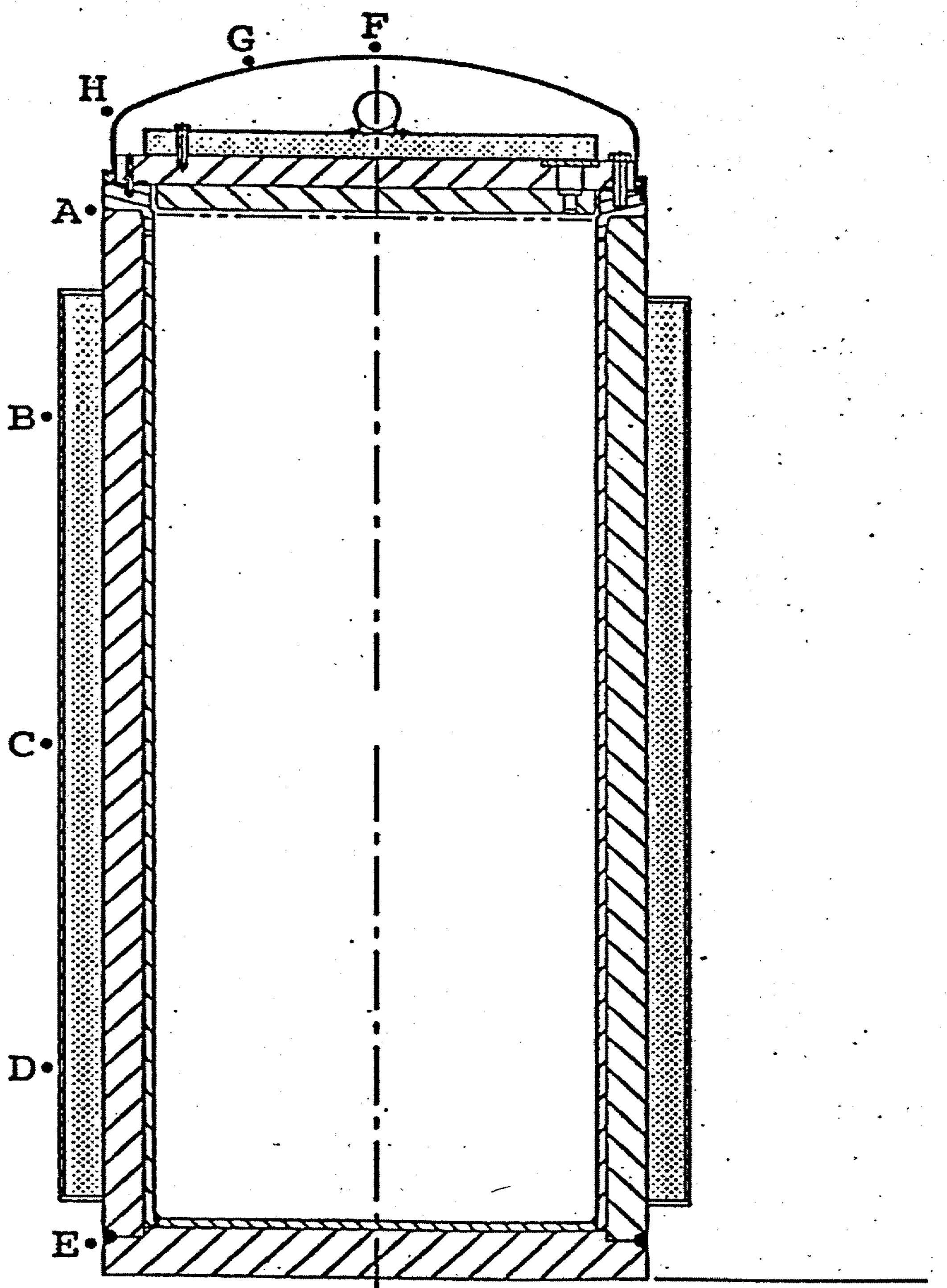
   c. Below Radial Neutron Shield (E): Midway between the bottom of the radial neutron shield and the bottom of the cask. At least six measurements equally spaced circumferentially.

   d. Top of Cask (F, G, and H): At the center of the protective cover, one measurement (F). Halfway between the center and the knuckle at least four measurements equally spaced circumferentially (G). At the knuckle at least four measurements equally spaced circumferentially (H).

8. The average dose rates shall be determined as follows.

   In each of the four measurement zones in TS 5.2.3.7, the sum of the dose rate measurements is divided by the number of measurements to determine the average for that zone. The neutron and gamma-ray dose rates are averaged separately. Uniformly spaced dose rate measurement locations are chosen such that each point in a given zone represents approximately the same surface area.

Figure 5.2.3-1
Contact Dose Rate Measurement Locations



TRANSNUCLEAR, INC.

TN-68 DRY STORAGE CASK SYSTEM

SAFETY EVALUATION REPORT

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LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BNFL	British Nuclear Fuels PLC
BPV	Boiler and Pressure Vessel Code
BTU	British Thermal Unit
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CVN	Charpy V-Notch
DBE	Design Basis Earthquake
DCF	Dose Conversion Factors
DCSS	Dry Cask Storage System
DLF	Dynamic Load Factor
DOE	Department of Energy
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
GE	General Electric
ISFSI	Independent Spent Fuel Storage Installation

ISG	Interim Staff Guidance
kW	Kilowatts
lbm	Pounds (Mass)
LCO	Limiting Conditon for Operation
LEFM	Linear Elastic Fracture Mechanics
LLNL	Lawrence Livermore National Laboratory
LST	Lowest Service Temperature
LWR	Light Water Reactor
MCNP	Monte Carlo Neutron Photon Code
M&TE	Measuring and Test Equipment
MWD/MTU	MegaWatt Days Per Metric Ton of Uranium
NDE	Nondestructive Examination
NQA-1	Nuclear Quality Assurance-1
NRC	Nuclear Regulatory Commission
OMS	Overpressure Monitoring System
ORNL	Oak Ridge National Laboratory
ppm	Parts Per Million
psia	Pounds Per Square Inch Absolute
psig	Pounds Per Square Inch Gauge
PT	Liquid Penetrant Examination
PWR	Pressurized Water Reactor
QA	Quality Assurance
QAP	Quality Assurance Program
RT	Radiographic Examination

SAR	Safety Analysis Report
SER	Safety Evaluation Report
SRP	Standard Review Plan
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
TEDE	Total Effective Dose Equivalent
TN	Transnuclear, Inc.
T_{NDT}	Nil Ductility Transition Temperature
TS	Technical Specifications
USL	Upper Subcritical Limit
UT	Ultrasonic Examination
VT	Visual Examination
WRC-107	Welding Research Council-107

INTRODUCTION

This Safety Evaluation Report (SER) documents the review and evaluation of Revision 5 to the Safety Analysis Report (SAR) for the Transnuclear, Inc. (TN) TN-68 Dry Storage Cask System[1]. The SAR, submitted by TN, follows the format of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems[2]. This SER uses essentially the same Section-level format, with some differences implemented for clarity and consistency.

The review of the SAR addresses the handling and dry storage of spent fuel in a single dry storage cask design, the TN-68. The cask would be used at an Independent Spent Fuel Storage Installation (ISFSI) that would be licensed under 10 CFR Part 72[3] at a reactor site operating with a 10 CFR Part 50 license.

The staff's assessment is based on whether the applicant meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection. Decommissioning, to the extent that it is treated in the SAR, presumes that, as a bounding case, the TN-68 cask is unloaded and subsequently decontaminated before disposition or disposal.

References

1. TN-68 Dry Storage Cask Safety Analysis Report, Rev. 5, Transnuclear Inc., May 1999.

2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

3. U.S. Code of Federal Regulations. "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.

1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the TN-68 dry storage cask system is to ensure that Transnuclear, Inc. has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the cask.

1.1 System Description and Operational Features

The TN-68 cask accommodates 68 intact boiling water reactor (BWR) fuel assemblies and consists of the following components (see Figure 1-1):

- A basket assembly which locates and supports the fuel assemblies.
- An inner confinement vessel (and lid) which comprises the primary confinement barrier.
- A carbon steel gamma shield structure surrounding the primary confinement vessel.
- Neutron shielding material (jacketed) exterior to the gamma shield.
- A protective cover which provides weather protection for the closure lid and seal components, the top neutron shield, and the overpressure system.
- An overpressure monitoring system which monitors pressure between the two seals of the cask lid. This system allows for early detection of cask seal leakage.
- Sets of upper and lower trunnions for lifting and support of the cask.

TN-68 casks are to be stored at a minimum of 16 ft apart, center to center.

1.2 Drawings

The drawings for the TN-68 associated with the structures, systems, and components (SSCs) important to safety are contained in Section 1.5 of the SAR[1]. A specific list of these components is noted on the parts list shown on drawing 972-70-2. The applicant provided sufficiently detailed drawings regarding dimensions, materials, and specifications to allow a thorough evaluation of the entire system. Specific SSCs are evaluated in Sections 3 through 14 of this SER.

1.3 Cask Contents

The approved contents for the TN-68 are specified in the Technical Specifications (TS). The TN-68 cask is designed to store up to 68 intact BWR fuel assemblies manufactured by General Electric (GE). The maximum allowable initial lattice-average enrichment of the fuel to be stored is 3.7 wt% U-235. A description of the fuel assemblies is provided in Section 2.1 of the SAR.

1.4 Qualification of the Applicant

TN provides the design, analysis, licensing support, and quality assurance (QA) for the TN-68 cask. Fabrication of the cask is done by one or more qualified fabricators under TN's QA program. Section 1.3 of the SAR adequately details TN's technical qualifications and previous experience in the area of dry storage cask licensing.

1.5 Quality Assurance

The quality assurance program (QAP) is evaluated in Section 13 of this SER.

1.6 Evaluation Findings

F1.1 A general description and discussion of the TN-68 is presented in Section 1 of the SAR (Rev 5), with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

F1.2 Drawings for SSCs important to safety are presented in Section 1 of the SAR. Specific SSCs are evaluated in Sections 3 through 14 of this SER.

F1.3 Specifications for the spent fuel to be stored in the dry storage cask system are provided in Section 2 of the TN-68 SAR.

F1.4 The technical qualifications of the applicant to engage in the proposed activities are identified in Section 1.3 of the SAR and are acceptable to the NRC staff.

F1.5 The QAP is described in Section 13 of the SAR and is evaluated in Section 13 of this SER.

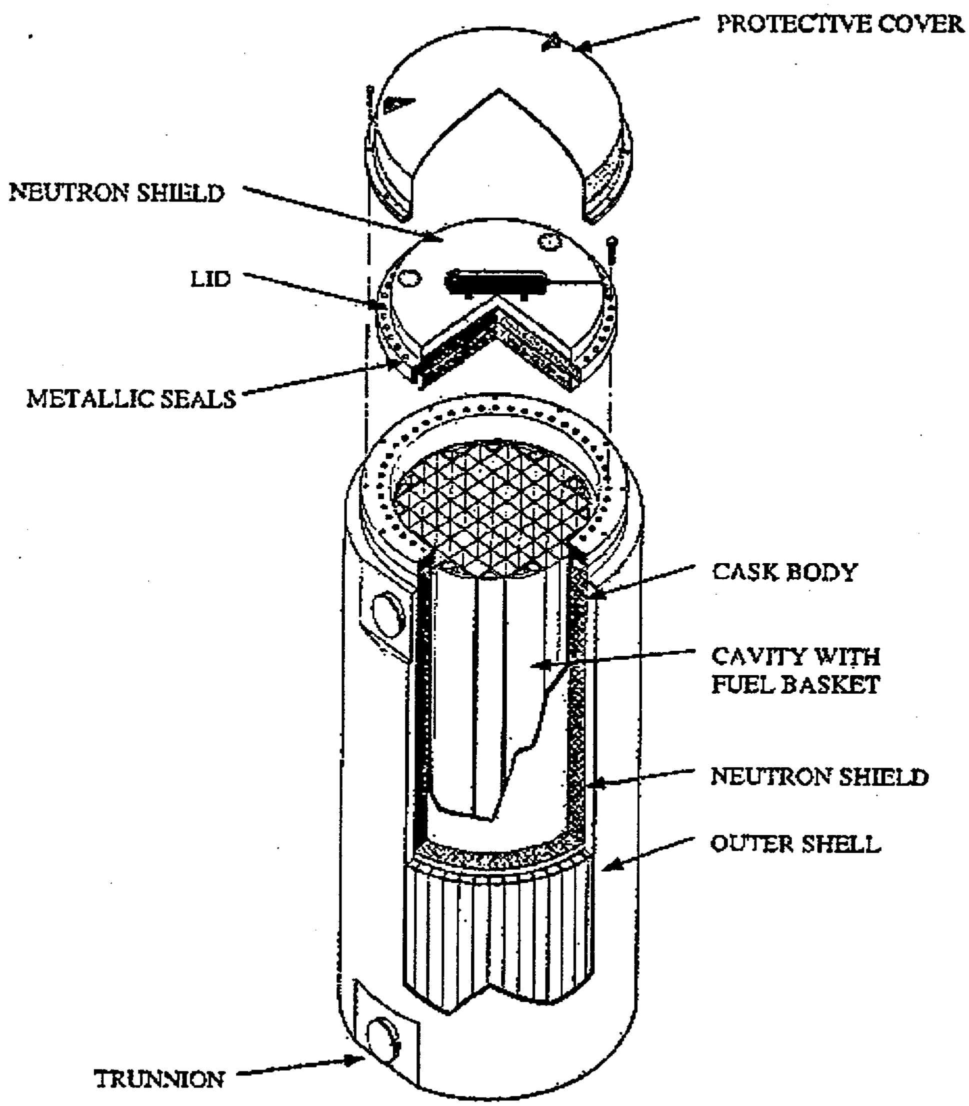
F1.6 The TN-68 cask was not reviewed in this SER for use as a transportation cask.

F1.7 The staff concludes that the information presented in Section 1 of the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is based on a review that considered the regulation itself, Regulatory Guide 3.61, and accepted dry storage cask practices detailed in NUREG-1536[2].

1.7 References

1. TN-68 Dry Storage Cask Safety Analysis Report, Rev. 5, Transnuclear Inc., May 1999.

2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.



TN-68 CASK

Figure 1-1 TN-68 Dry Storage Cask

2.0 PRINCIPAL DESIGN CRITERIA

The objective of evaluating the TN-68 principal design criteria applicable to the SSCs important to safety is to ensure that they comply with the relevant general criteria in 10 CFR Part 72.

The design criteria listed in Revision 5 of the TN-68 SAR[1] was reviewed and a detailed evaluation is given in Sections 3 through 14 of this SER.

2.1 Structures, Systems, and Components Important to Safety

Table 2.3-1 and Drawing 972-70-2 of the SAR identify the cask SSCs important to safety. For the cask components classified as not important to safety, TN provided justification for their exclusion in Section 2.3 of the SAR.

2.2 Design Bases for Structures, Systems, and Components Important to Safety

The TN-68 design criteria summary includes the allowed range of spent fuel configurations and characteristics, the enveloping conditions of use, and the bounding site characteristics.

2.2.1 Spent Fuel Specifications

The TN-68 is designed to store 68 intact, unconsolidated, General Electric (GE) boiling water reactor (BWR) spent fuel assemblies with or without fuel channels. Section 2 of the SAR provides detailed fuel assembly parameters which includes the fuel type, assembly and uranium mass for the design basis assembly, enrichment, burnup, and cooling time. This section of the SAR also specifies the bounding fuel types for the criticality, shielding, thermal, and confinement analyses within the SAR.

The fuel characteristic limits are given in TS 2.1. These limits are based on the criticality, shielding, thermal, and confinement analyses which are evaluated in Sections 3 through 14 of this SER.

2.2.2 External Conditions

Section 2.2 of the SAR identifies the bounding site environmental conditions and natural phenomena for which the TN-68 is analyzed. These are evaluated in Sections 3 through 14 of this SER. TS 4.3 identifies the bounding site-specific parameters for the TN-68.

Sections 2 and 11 of the SAR identify the normal, off-normal, and accident conditions evaluated. The staff's evaluation of the TN-68 response to the off-normal and accident conditions is in Section 11 of this SER.

2.3 Design Criteria for Safety Protection Systems

2.3.1 General

Section 2 of the SAR states that the minimum design life of the TN-68 is 40 years. The material mechanical properties analysis in Section 3.3 of the SAR is for a design life of 40 years. This design life bounds the 20-year period for the cask certificate.

The codes and standards of design and construction are specified in Sections 2.5, 3, and 7 of the SAR. Justification for exceptions to codes and standards is given in TS Table 4.1-1. SSCs important tc safety are designed, fabricated, and tested to quality standards which conform to the criteria of 10 CFR Part 72.

The TN-68 has an overpressure monitoring system which meets the intent of the continuous monitoring requirement of 10 CFR Part 72. This is evaluated in Section 7 of this SER.

2.3.2 Structural

Section 3 of the SER evaluates the structural integrity of the TN-68 under the combined normal, off-normal, and accident loads. Loading combinations are classified as Service Conditions, consistent with Section III of the ASME Boiler and Pressure Vessel Code[2], and the resulting stresses are evaluated. The TN-68 structural components are designed to protect the cask contents from significant structural degradation, preserve retrievability, and maintain subcriticality, and confinement.

2.3.3 Thermal

Section 4 of this SER evaluates the TN-68 thermal design criteria. Normal condition thermal design criteria include maintaining the integrity of confinement, fuel cladding, and maintaining the neutron shield. The TN-68 is designed to passively reject decay heat, and the heat removal mechanisms are independent of intervening actions under normal and off-normal conditions.

2.3.4 Shielding/Confinement/Radiation Protection

Sections 5, 7, and 10 of this SER evaluate the TN-68 design criteria which protects occupational workers and members of the public against direct radiation and radioactive material releases, and which minimizes doses after any postulated off-normal or accident condition, sufficient to meet the requirements of 10 CFR Part 72. Section 11 of this SER evaluates the effect of radiological consequences for hypothetical accidents. The TN-68 uses a bolted lid closure system, double metallic lid and lid penetration seals, and a combined cover-seal pressure monitoring system to provide confinement. Radiation exposure is minimized by the neutron and gamma shields and by operational procedures.

2.3.5 Criticality

The TN-68 has been designed to assure that the effective neutron multiplication factor is less than or equal to 0.95 under all credible conditions. Section 6 of this SER evaluates the control methods which maintain the subcriticality of the system. The control methods used include a neutron absorbing material in the basket and a minimum basket cell opening. The continued efficacy of the neutron absorber plates over a 20-year storage period is assured by the design of the TN-68 cask. The neutron flux in the dry cask over the storage period is also very low such that depletion of the Boron-10 in the neutron absorber is negligible.

2.3.6 Cask Operations

Cask operations, which are evaluated in Section 8 of this SER, include descriptions of generic procedures for loading and unloading. Radiation protection features, including features to facilitate decontamination, are incorporated in both the physical design and the operating procedure descriptions.

2.3.7 Acceptance Tests and Maintenance

TN-68 acceptance tests and maintenance programs are evaluated in Section 9 of this SER.

2.3.8 Decommissioning

The TN-68 decommissioning considerations are presented in Sections 2.4 and 14 of the SAR and evaluated in Section 14 of this SER.

2.4 Review Summary

TN presented general details of the principal design criteria in Section 2 of the SAR and provided appropriate details in the associated Sections of the SAR.

2.5 Evaluation Findings

F2.1 The staff concludes that the principal design criteria of Revision 5 of the TN-68 SAR are acceptable with regard to demonstrating compliance with the regulatory requirements of 10 CFR Part 72. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and acceptable engineering practices. More detailed evaluations of the design criteria and assessments of compliance with those criteria are presented in Sections 3 through 14 of this SER.

2.6 References

1. TN-68 Dry Storage Cask Safety Analysis Report, Rev. 5, Transnuclear Inc., May 1999.

2. ASME Boiler and Pressure Vessel Code, Section III, Division I, 1995 including 1996 addenda.

3.0 STRUCTURAL EVALUATION

This section evaluates the structural design of the TN-68 cask. Structural design features and design criteria are reviewed, and analyses related to structural performance under normal, off-normal, accident, and natural phenomena events are evaluated.

The scope of this evaluation includes all structural material presented in the TN-68 SAR[1]. This included Section 3 of the SAR, Appendix 6A (fuel response during cask impact), and the natural phenomena material in Section 2.

Loads and load combinations are reviewed for the normal, off-normal, accident, and natural phenomena events categorized in NUREG-1536[2]. Structural material specifications are reviewed and compared with acceptable codes and standards. Design assumptions and analytical approaches are reviewed for appropriateness and acceptability. Critical stresses and the construction of the TN-68 cask components are reviewed to ensure they meet the design codes and standards acceptance criteria.

3.1 Structural Design

3.1.1 Structural Design Features

The TN-68 cask consists of the following four components:

1. Confinement boundary
2. Non-confinement boundary
3. Fuel Basket
4. Trunnions

The confinement boundary of the TN-68, identified in Figure 1.2-1 of the TN-68 SAR, consists of the inner shell (both the cylindrical portion as well as the bottom plate), the closure flange out to the seal seating surface, and the lid assembly outer plate. The lid bolts and seals are also part of the confinement boundary.

The non-confinement boundary components consist of the gamma shielding, neutron shield outer shell, and trunnions. While these components do not have a confinement function, they must react to the confinement or environmental loads, and in some cases, share loading with the confinement components.

The classification of components as "important to safety" and "not important to safety" is contained in Section 2 and summarized in Table 2.3-1 of the SAR. These components are annotated on Drawing 972-70-2 located in Section 1 of the SAR. Components considered important to safety include the confinement vessel, lid bolts and gasket, lid vent and drain covers and bolting, basket assembly, trunnions, and radial neutron shield. Items considered not important to safety include the drain tube, Hansen couplings, pressure monitoring system, protective cover, basket key, fuel spacers, basket rail shims, security wire and seals, lid alignment pin, top neutron shield, and shield ring.

3.1.2 Structural Design Criteria

The TN-68 SAR uses several design criteria to ensure that the cask design meets the requirements of 10 CFR Part 72. Section 3.1 of the SAR describes the design criteria for the four cask components listed in Section 3.1.1 above.

The main design code/standard used for the TN-68 cask design is the ASME Boiler and Pressure Vessel (BPV) Code[3]. The particular ASME portion used is Section III, Division 1, Subsection NB (1995 edition). This criterion is used to the maximum practical extent, and exceptions to strict usage are given in Section 7 of the SAR and Table 4.1-1 of the TS. The TN-68 cask is not code stamped. The QA requirements of Nuclear Quality Assurance-1 (NQA-1) or 10 CFR Part 72 Subpart G are impose I in lieu of specific provisions dealing with fabricator qualifications, etc., usually covered by Subsection NCA of Section III, Division 1. Load conditions are categorized as either normal or hypothetical; the former category being compared to Level A service limits and the latter being compared to Level D service limits. It is noted by the staff that the use of the ASME Section III, Division 1, Subsection NB application very closely resembles the more recent ASME Section III, Division 3 which is specifically designed to address nuclear packaging. In summary, the staff agrees that the following elements of the ASME code are utilized in the design:

1. Material selection and certification.
2. Allowable material stress values.
3. Stress categorization procedures (membrane, membrane plus bending, etc.).
4. Selection of weld types and weld inspection procedures.

3.1.2.1 Individual Loads

Load conditions for both normal and off-normal events are described in Sections 2 and 3 of the SAR. Summary data is presented in SAR Tables 2.2-5 through 2.2-9. Rather than grouping the loading as normal, off-normal, and accident-level, the loads were designated either normal (Level A) or accident (Level D). Some cask components under off-normal conditions conservatively use the same allowable stresses as for normal conditions even though the relevant ASME allowable stresses are greater.

Normal loading for the cask and contents is described in Tables 2.2-5 and 2.2-6 of the SAR. It should be noted that loads specified as design loads for the cask represent a subset of those categorized as Level A, or normal loads. The structurally significant normal loading conditions are primarily loadings due to internal and external pressure and lifting loads. Other loads result in only minor structural effect.

Load conditions categorized as off-normal and accident-level in NUREG-1536 are grouped as Level D loadings and are summarized in SAR Table 2.2-7. It is noted that the fire accident case is omitted from Table 2.2-7; however, fire accident is analyzed in Section 3 of the SAR. In addition, explosive overpressure is omitted from the Table 2.2-7 and Section 3. This is addressed in Section 3.1.2.1.3 below. All other accident-level natural phenomena load cases listed in NUREG-1536 are listed and treated either in Sections 2 or 3 of the SAR.

SAR Tables 2.2-8 and 2.2-9 list normal and accident load cases and the series of combinations in which the stress levels for the individual load cases are combined. It is noted by the staff that thermal stresses are omitted from Table 2.2-9 (accident conditions) but are adequately evaluated in Section 3.4.6 of the SAR.

3.1.2.1.1 Tipover

The TN-68 cask will not tipover as a result of a postulated natural phenomenon event, including tornado wind, tornado-generated missile, seismic event, or flood. To demonstrate the defense-in-depth features of the design, a non-mechanistic tipover scenario is analyzed. Section 3.3.1 below discusses the tipover analysis performed in the SAR.

3.1.2.1.2 Handling Accident

Handling accidents for the TN-68 cask are considered to be side and end drop events. These are evaluated in Section 3.3 below.

3.1.2.1.3 Explosive Overpressure

Explosive overpressure is not addressed in either Sections 2 or 3 of the SAR. The cask is designed to withstand an external pressure of 25 psi as described in Section 2.2.5.3.4 of the SAR. If a credible explosion is identified that would apply more than 25 psi to the outer surface of the cask at a site, the site will have to address this issue in its 10 CFR 72.212 evaluation.

3.1.2.1.4 Flood

Flood loading is addressed in Section 2 of the SAR. The TN-68 cask is evaluated for a water depth of 57 ft. Drag force (due to flood) evaluation is performed in Section 2.2.2.3 of the SAR. This evaluation demonstrated that for anticipated friction coefficients, water velocity of 22.1ft/sec would be required to cause cask motion.

3.1.2.1.5 Tornado and Tornado Missile

Tornado and tornado missile loadings are addressed in Section 2 of the SAR. The TN-68 cask is evaluated for a design-basis tornado wind velocity of 360 mph and a pressure drop of 3 psi. Tornado missiles are listed in Section 2.2 of the SAR. Stability of the TN-68 cask due to tornado missile impact is evaluated in Section 3.4.4 below.

3.1.2.1.6 Earthquake

The design earthquake for use in the design of an ISFSI must be equivalent to the safe shutdown earthquake (SSE) for the nuclear power plant, the site of which has been evaluated under the criteria of 10 CFR Part 100, Appendix A. The TN-68 cask is evaluated for an applied horizontal acceleration of 0.26g and a vertical acceleration of 0.17g. These earthquake inertia forces are assumed to be applied at the top of the concrete pad. Section 3.4.2 below evaluates cask seismic response.

3.1.2.1.7 Snow and Ice

Snow and ice loadings are addressed in Section 2.2.4 of the SAR. Section 3.4.5 below evaluates snow and ice loadings corresponding to 50 psf on the TN-68 cask.

3.1.2.1.8 Lightning

The effects of lightning on the cask are addressed in Section 2.2.5.2.8 of the SAR. Lightning will not cause a significant thermal effect. If struck by lightning on the lid, the electrical charge will be conducted by paths provided by the lid bolts to the body. Due to the massive size of the cask body and the highly conductive carbon steel construction, the staff concludes that lightning would not pose a structural concern for the TN-68 cask.

3.1.2.1.9 Fire

Temperatures from the thermal analysis of a fire event performed in Section 4 of the SAR are utilized in Appendix 3A to evaluate the thermal stress response of the cask. These stress values are reported in Tables 3A.2.3-9 and 3A.2.3-10 of the SAR. Due to the low values of stress, the staff concurs that thermal stress effects of the fire are acceptable.

3.1.2.2 Loading Combinations

Loading combinations used in the SAR are listed in Table 2.2-8 for normal conditions and Table 2.2-9 for accident conditions. The staff agrees that these load combinations simulate the structural events modeled.

3.1.2.3 Allowable Stresses

Allowable stress values for the various cask materials are listed in SAR Tables 3.3-1 and 3.3-4. The staff concludes that these values meet the ASME allowable stresses, based on the appropriate ASME subsections and service levels, and that appropriate considerations to elevated thermal effects were given.

3.1.3 Weights and Center of Gravity

The gross weight of the TN-68 cask and contents is approximately 230,000 lbs. The center of gravity of the cask is located on the axial centerline 97.22 inches from the base of the cask. Weights and locations of the center of gravity (measured along the axial centerline from the base of the cask) of various cask components are listed in SAR Table 3.2-1. A conservatively high weight is used for the structural analyses. A conservatively low weight and high center of gravity are used for the analysis of the stability of the cask.

3.1.4 Materials

The structural materials used for the TN-68 are listed in Section 3 of the SAR. Table 3.3-6 of the SAR lists the primary function of each cask component along with information on drawing number, if applicable; safety class, codes/standards (including welding); coatings; and pertinent service conditions such as stress, temperature, time, pressure, environment, and important mechanical properties. In addition, Section 3.3 of the SAR discusses mechanical properties of materials and other tabulations of pertinent mechanical properties.

Material properties are generally taken from Section III, Part D of the ASME BPV code (1995 editions) when possible. These materials are either Class 1, 2, or 3 materials, or they do not belong to these classes. Materials other than ASME Code materials are permitted as discussed in NUREG-3854, Fabrication Criteria for Shipping Containers, and NUREG-1617, Standard Review Plan for Transportation Packages for Spent Nuclear Fuel, for the fabrication of casks. The materials used for the cask body (gamma shield, confinement shell, bottom, and top) are various grades of carbon steels and are described on page 3.1-1 of the SAR.

The confinement shell and bottom plate are designed with SA 203 Grade E material. The lid material selected is either SA-350 Grade LF3 or SA 203 Grade E. Both steels are Section III, Class 1 materials.

The cylindrical gamma shield shell is SA-266 Class 2 material. The gamma shield cylindrical shell plate is SA-266 Class 2 or SA-516 Grade 70, and the bottom shield plate is fabricated from either SA-105 or SA-516 Grade 70. All three of these steels are Section III, Class 1 materials.

Materials of the TN-68 fuel basket are described in Section 3.1.2.3 of the SAR. No structural consideration is given to the potential load carried by the basket's poison plates (either borated aluminum, or a boron carbide / aluminum metal matrix composite). Structural detail of basket fabrication is given on drawings 972-70-4 and 972-70-5 in the SAR. Axial support rails support the basket at the inside cask wall. These are composed of 6061-T6 aluminum. Load-bearing materials are the aluminum basket rails (6061-T6), the stainless steel square tubing and stainless steel plate (SA-240 Type 304). The basis for the allowable stress for the Type 304 stainless steel fuel-compartment box, plate and 6061-T6 alloy is Section III of the ASME Code.

The trunnion material for the TN-68 design is SA-105. This material is listed as ASME Class 1 material. SAR Table 3.3-2 gives allowable trunnion stress values taken from Table 2A of ASME Section II Part D.

The fracture toughness of ferrous components is assessed in Appendix 3E of the SAR. This is done in the process of determining pre-service and in-service inspection requirements and allowable flaw sizes for various loading conditions and temperatures (see Section 3.1.4.4 below).

3.1.4.1 Material Compatibility and Durability

Compatibility of materials used in fabrication of the TN-68 is addressed in Section 3.4.1 of the SAR, which reviews chemical, galvanic, and other interactions between the materials and contents for the environmental conditions encountered during loading, storage, handling, and unloading. Discussions of environmental conditions associated with each phase of service are presented in Section 3.4.1 of the SAR. System components are expected to have excellent corrosion resistance, compatibility with one another, and durability in their respective environments. Periodic maintenance will be done, as needed, on external coatings. A seal replacement can be accomplished if needed during the lifetime of the system.

The Type 304 stainless steel components and welds of the interior basket assembly are not expected to be significantly affected by the adverse presence of either deionized water or the other environmental conditions of temperature and time under service conditions. The staff concurs that there is no significant chemical reaction between the stainless steel plates, the aluminum rails, and the borated aluminum or boron carbide aluminum composite plates used for criticality control. In addition, cask operation descriptions are specified to preclude excessive build up of hydrogen when water is inside the cask for periods well beyond those expected.

There is one potential exception to the lifetime corrosion resistance of the TN-68 system. Corrosion could occur at the crevice formed where the outer metallic seal contacts the sealing surface. The moisture necessary for this crevice corrosion to occur is not likely to be present because the combined effects of the weather cover and the decay heat from the stored fuel will maintain a low humidity at this seal. Staff notes that this seal material has a very good record of performance and endurance and is not expected to fail during the licensed life of the system. If this seal were to somehow fail during this period, there would be no safety significance as the failure would automatically be detected by the pressure monitoring system before any adverse effects related to the cask function would occur. A replacement seal would be installed for continued service.

The staff concurs that the factors which have a potential for affecting service performance of components, e.g. chemical reactions, galvanic reactions, thermal radiation effects, or other reactions and interactions between materials and the environment, are not likely to adversely affect the material properties during handling and storage operations of the 20-year licensed service period of the cask.

3.1.4.2 Welds

Weld specification and inspection techniques are discussed in Section 3.1.1 of the SAR. Various standards of the ASME Boiler and Pressure Vessel Code are applied. Confinement boundary weld types are in accordance with Section III, Subsection NB. Acceptance standards are those of Article NB-5000. Test standards are in accordance with Section V, welding standards are in accordance with Section IX, and materials are in accordance with Section II, Part C. The staff concurs that sufficient detail to welding has been given in the cask design.

3.1.4.3 Bolting Materials

Bolting materials used in the TN-68 cask are listed on Drawing 972-70-2 in the SAR. All bolting is listed ASME Section II Part D, Table 4, indicating ASME Class 1 compliance. The bolting materials are:

Lid Bolts: Protective Cover Bolts, Top Neutron Shield Bolts, Drain ,Vent, and OverPressure Port	SA-540, Gr. B24 Class 1
Cover bolts:	SA-193, Gr. B-7
Trunnion bolts:	SA-320, Gr. L43

3.1.4.4 Brittle Fracture of Materials

Fracture toughness of the TN-68 cask confinement boundary, gamma shield, and welds is addressed in Appendix 3E of the SAR. Each is evaluated below.

The TN-68 cask is designed for ambient temperatures as low as –20°F. The confinement boundary materials (including lid bolts) are selected to meet the fracture toughness criteria of ASME Code, Section III, Division 3, Subsection WB. While the cask is designed to meet NB requirements, it is noted that the use of WB requirements for fracture toughness indicates that both the requirements of NB and the two brittle fracture Regulatory Guides recommended in NUREG-1536 have been met. ASME Table WB-2331.2-1 of Section III, Subsection WB (Para. WB-2330) is used to determining the nil ductility transition temperature (T_{NDT}) of the confinement boundary design. The results indicate that T_{NDT} for the 1.5-inch thick confinement shell and bottom plate is -80°F; T_{NDT} for the 7.5-inch thick flange is -133°F; and T_{NDT} for the 5-inch thick lid plate is -126°F. In addition, Charpy V-Notch (CVN) testing of the confinement boundary materials will also be conducted at a temperature no greater than 60°F above the T_{NDT} to ensure that they will not be susceptible to brittle fracture at –20°F. The acceptance criteria is 35 mil lateral expansion and 50 ft-lb. absorbed energy. The fracture toughness requirements of the lid bolts will meet the criteria of ASME Code, Section III, Division 3, Subsection WB (Para. WB-2333). CVN testing will be performed at –20°F. The acceptance criteria is that the material exhibit at least 25 mils lateral expansion as per ASME Table WB-2333-1. Confinement boundary welds will be examined by radiographic and either liquid penetrant or magnetic particle examinations in accordance with Section III, Subsection NB, Paragraphs NB-5210, NB-5220, and NB-5230.

The gamma shield is not part of the confinement boundary, however, it provides structural support to the confinement boundary during drop accidents. Because of the TN-68 cask design geometry, cracks in the gamma shield will not propagate into the confinement boundary. The gamma shield will not separate from the confinement boundary, due to the frictional forces between the confinement vessel and the gamma shield which arise as a result of a shrink fit of the gamma shield shell over the confinement shell. The allowable flaw sizes in the gamma shield design are calculated using a linear elastic fracture mechanics (LEFM) methodology, from Section XI of the ASME Code (1989 editions). The results of the fracture toughness analysis indicate that the critical flaw sizes (flaws large enough to give rise to rapid unstable extension) are larger than those typically observed in forged steel and plate components' flaws, which is true for flaws either in the gamma shield shell or in the top or bottom shield plates. Therefore, no special examination is required of the gamma shield to ensure the absence of flaws that would result in unstable crack growth or brittle fracture.

If the bottom plate weld were to fail, the bottom plate could become detached, which would impact the shielding capability of the cask. At –20°F, LEFM analysis indicates that the minimum allowable flaw sizes for surface and subsurface are 0.27 in. and 0.40 in., respectively.

The following inspections (made prior to placing the cask in service) are required to ensure that large defects (those equal to or larger than the above flaw sizes) are detected and repaired:

1. liquid penetrant or magnetic particle test at base metal
2. liquid penetrant or magnetic particle test at root pass
3. liquid penetrant or magnetic particle test for each 0.375 -inches of weld
4. liquid penetrant or magnetic particle test at final surface

Failure of the weld between the gamma shield and top flange would not have any safety significance. The gamma shield will not separate from the confinement boundary. Therefore, only liquid penetrant or magnetic particle tests of the final surface are specified. Failure of the weld between the top shield plate and lid could result in a drop of the top shield plate into the cask cavity, although the top shield plate will still remain inside the confinement boundary due to the cask arrangement and would not lose its shielding capability. The inspection requirements specified for this location are the same as that specified for the bottom plate weld above.

The liquid penetrant or magnetic particle examinations will be in accordance with Section V, Article 6 of ASME Code.

3.1.4.5 Materials Conclusion

The staff concludes that the materials of construction as specified in the TN-68 cask design are adequate for meeting the service requirements and for performing all safety functions including the structural, thermal, shielding, criticality, and confinement functions.

3.1.5 General Standards for Cask

The structural analyses for the cask must ensure positive closure, adequate safety factors for lifting devices, and that there is no adverse effect to the safe storage of the spent fuel due to chemical or galvanic reactions. The most important function of structural analyses is to show sufficient structural capability of the TN-68 system to withstand the postulated worst-case loads under normal, off-normal, accident, and natural phenomena events with adequate margins of safety to preclude the following consequences:

1. unacceptable risk of criticality,
2. unacceptable release of radioactive materials,
3. unacceptable level of radiation, and
4. impairment of retrievability.

The structural analyses presented in Section 3 of the SAR demonstrate that the cask will maintain confinement during normal and off-normal operations, accident conditions, and natural phenomena events. Section 2 of the SAR justifies that the cask will maintain confinement for natural phenomena events. In addition, results from Appendices 3A, 3B, and 3C of the SAR indicate that gross ruptures will not occur in the fuel cladding during accident conditions. In Appendix 3D of the SAR, a finite element model is used to determine cask response due to a tipover event. The results from that effort demonstrate that fuel damage sufficient to cause retrieval concerns will not occur during tipover.

Normal, off-normal, accident, and natural phenomena loading will not be sufficiently severe to cause degradation of the gamma shield performance. However, the neutron shield may be damaged by either tornado Missile A or B as defined in the SAR. Radiological effects due to a loss of the neutron shield are addressed in Section 10 of this SER.

The above-mentioned SAR analyses are evaluated in Sections 3.1.4, 3.2, 3.3, and 3.4 of this SER.

3.2 Normal Operating and Off-Normal Conditions

3.2.1 Chemical and Galvanic Reactions

Discussion of potential chemical and galvanic reactions is given in SAR Section 3.4.1. In this SER, these reactions are discussed with other considerations in Section 3.1.4, especially Section 3.1.4.1 on material compatibility and durability. The staff concurs that such reactions have been sufficiently addressed in the design and do not adversely affect cask performance.

3.2.2 Positive Closure

The TN-68 cask lid is bolted directly to the upper ring forging. Access to the lid requires removal of the protective cover. Deliberate loosening of bolts requires extensive effort, using appropriate equipment. The large preload applied to lid bolts prevents inadvertent opening of the cask closure lid from loads such as bottom-end drop and thermal expansion. In addition, two of the bolts for the protective cover are installed with security seals to ensure that unauthorized closure access will be detected. Therefore, the TN-68 cask cannot be opened unintentionally.

3.2.3 Lifting Devices Analysis

The TN-68 cask design has two upper trunnions and two lower trunnions. The upper trunnions are the only components used for lifting. Each upper trunnion is attached to the gamma shield by twelve 1.5-inch 8UN-2A bolts. The lower trunnions are used to rotate the cask from horizontal orientation to the vertical orientation and are welded to the gamma shield .

Structural effects due to lifting loads passed from the trunnion to the gamma shield are analyzed in Section 3.4.3.1 of the SAR. In accordance with ANSI N14.6[4], the single failure proof upper trunnions of the TN-68 cask are designed with a safety factor of 6 against the trunnion material yield stress and 10 against the trunnion material ultimate stress. Although the gross weight of the TN-68 package is approximately 229,000 lbs., a cask weight of 240,000 lbs. is used in the trunnion analysis. In addition, a dynamic amplification factor of 1.15 is also used on the dead weight load. Stresses in the trunnion material are analyzed using beam shear and bending calculations at several cross-sections of the trunnion. Stresses in the upper trunnion flange bolts are analyzed using conventional textbook methods and are presented in Section 3.4.3.3 of the SAR. The results indicate that calculated stresses are below the allowable stresses. Therefore, the TN-68 trunnion design is adequate.

Stress concentrations caused by the trunnion loads acting on the gamma shield are analyzed using the techniques of Welding Research Council-107 (WRC-107)[5]. These local stresses are superimposed on the stresses of the ANSYS[6] cask body model of SAR Appendix 3A in arriving at the stress values used in evaluations.

3.2.4 Pressure and Temperature Effects

Stress levels in the cask body due to pressure and thermal loads are evaluated in Appendix 3A of the SAR. The design internal pressure of 100 psi is used for these evaluations. Chapter 7 of the SAR shows that this is substantially in excess of maximum anticipated cavity pressure even under accident conditions. Temperatures used for thermal stress evaluation are summarized in Table 3.3-3 of the SAR. The staff notes that the temperatures used in the analysis are substantially greater than the maximum calculated corresponding component temperatures in Chapter 4 of the SAR.

Finite element modeling is performed using an axisymmetric model to assess stresses due to pressure and temperatures. Temperatures used are taken from thermal models discussed in Section 4 of the SAR. ASME code checks are performed in accordance with Level A load conditions of Subsection NB (normal conditions) and accident conditions. These results are summarized in Table 3.4-5 of the SAR for the confinement vessel and in Table 3.4-6 of the SAR for the gamma shield shell. In addition, the lid bolt stress levels were evaluated in Appendix 3A for both normal and hypothetical conditions. The lid bolt stress levels found in Appendix 3A of the SAR are summarized in Table 3.4-7 of the SAR. In each of these three tables, ASME code allowable stress levels for each case are presented. It is seen that large safety margins exist relative to ASME code allowable stresses.

In view of the conservative values of temperature and pressure used in these calculations and the large safety margins resulting, the staff concludes that structural effects of temperature and pressure loading have been adequately addressed and are acceptable.

3.3 Accident Conditions

3.3.1 Cask Tipover and Side Drop

The tipover analysis of the TN-68 cask is provided in Appendix 3D of the SAR. The methodology used in performing the analysis was developed by the Lawrence Livermore National Laboratory (LLNL)[7]. This methodology was verified by LLNL through comparison of analyses results with test data.

The TN-68 cask is conservatively assumed to be a rigid body. The peak rigid body accelerations of the TN-68 cask due to a tipover accident are predicted analytically using the LS-DYNA3D[8] finite element program. The TN-68 finite element model is made up of four components: the cask body, cask internals, concrete, and soil. Essential parameters of the four components are listed in Section 3D.3.2 of the SAR. The finite element models of the cask body and the cask internals are developed in a similar manner to the model represented in Reference 7. Cask features such as the trunnions, neutron shield, and protective cover are neglected in terms of stiffness, but their weight is lumped into the cask body density. Mesh sizes of the cask, basket, concrete, and soils are in reasonable agreement with those in Reference 7. Contact elements are used between the cask and concrete pad and between the concrete pad and the soil. The result of the analysis indicates that the TN-68 cask has a peak deceleration of 65g at the top end of the cask. Based on the comparisons of analytical results with results of experimental tests presented in Reference 7, the staff concurs that the applicant has adequately validated the finite element modeling technique and the LS-DYNA3D finite element program.

Although the TN-68 cask and contents will only be lifted in a vertical orientation, side drop loading is still included in the cask analysis. The 65g peak deceleration from the tipover analysis discussed above is used for an equivalent side drop analysis. During a tipover accident, the deceleration varies according to the distance from the center of rotation. Thus, along the axial length of the TN-68 cask, the minimum deceleration (0g) would occur at the bottom-end and the maximum deceleration (65g) would occur at the top surface of the lid. This corresponds to a 33g uniform load along the axial length. The equivalent side drop analysis assumes a uniform 65g load along the entire length of the TN-68 cask. In view of the above, the staff agrees that the equivalent side drop analysis will envelop the tipover accident event. Furthermore, the tipover analysis neglects the outer shell and aluminum boxes. These components will deform and absorb energy during the tipover accident. Therefore, the actual deceleration for tipover would be less than the 65g peak deceleration calculated above.

The maximum stress intensity in the cask body due to load combinations which include the 65g uniform load along the entire length of the TN-68 cask is calculated to be about 56ksi. As an indication of the degree of conservatism inherent in assumptions used, the SAR presents a simulation of tipover based on the triangular loading discussed above. This resulted in a peak cask body stress level of only about 34 ksi (Figure 3D.5-3 of the SAR).

Due to the use of bounding "g" load in the quasi-static analysis of cask body stresses during tipover and in view of the large safety margins on stress due to hypothetical (tipover) loading shown in Tables 3.4-5 and 3.4-6 of the SAR, the staff concludes that the cask body is adequately designed to withstand tipover events.

Since the basket is not modeled in detail in the LS-DYNA3D finite element program, it is necessary to transfer the loads from the finite element model used in the tipover analysis to the detailed model of the basket. The peak cask body deceleration (at the top surface of the lid) for tipover is scaled to get the peak g level at the top of the basket. This resulted in 58g's. Based on the acceleration magnitude, duration and shape (Section 3D.5.2 of the SAR), a dynamic amplification factor of 1.32 is calculated. The load on the basket as a result of tipover is modeled as a steady-state acceleration equal to 77g (58 x 1.32 = 77). This load is then conservatively applied uniformly in the transverse direction to the basket. Basket response due to the 77g tipover load is evaluated in Section 3.3.4 below.

3.3.2 Cask Bottom-End Vertical Drop

The cask bottom-end drop analysis of the TN-68 cask is provided in Section 3D.7 of the SAR. The analysis is based on the methodology of EPRI NP-4830[9] and EPRI NP-7551[10]. This methodology has been "benchmarked" by scale model drop testing at Sandia National Laboratories and full scale cask drop testing in England. The results[12] indicate that end drop tests have excellent correlation with those end drop results predicted by the EPRI methodology.

The TN-68 cask is assumed to be a rigid body in the EPRI methodology. The storage pad properties and the cask geometry are used to determine the pad hardness parameter. The EPRI reports give curves that show the force on the cask as a function of storage pad hardness. The concrete storage pad (backed by fill) used in the 18-inch bottom-end drop analysis is 3-feet thick with 6,000 psi concrete compressive strength. The yield strength of the steel reinforcement in the storage pad is 60,000 psi. The storage pad is backed by soil with soil modulus of 32,600 psi. Although the TN-68 cask drop height is 18 inches, the TN-68 SAR conservatively used the curve for a 20-inch drop height to yield a "g" level of 39g's. Because this "g" load is based on the assumption that the cask is rigid, Reference 11 recommends a dynamic load factor of 1.5 be used to account for actual cask primary mode of response. Therefore, the "g" level for the TN-68 cask bottom-end drop is further increased to 58.5g (39 x1.5 = 58.5). The cask and basket structural analyses are performed by using 60g for the 18-inch cask bottom-end drop.

Because the above conservative assumptions are used to determine the 18-inch cask bottom-end drop "g" level, the staff assess that 60 "g's" represents a bounding value for the bottom-end drop structural loading. The stress analysis presented in Appendix 3A of the SAR indicates that the maximum stress intensities in the cask body due to any load combination which includes the 60g bottom-end drop are well below the stress limits for the accident conditions. Consequently, the staff concludes that the cask body is adequately designed to withstand the bottom-end drop events.

The response of the fuel basket under the 60g end drop loads is evaluated in Section 3.3.4 below.

3.3.3 Cask Lid Bolt Analysis for Cask Impact

The TN-68 lid bolt design uses forty-eight (48) 1.875-inch diameter steel bolts. The stress analysis of the lid bolts is based on the methodology of NUREG/CR-6007[12]. The details of the analysis are provided in Section 3A.3 of the SAR. Bolt preload is selected to resist a maximum internal pressure in the cask cavity of 100 psi plus any dynamic loading such as those for the hypothetical bottom-end drop and tipover onto the concrete storage pad. Quasi-static analyses are performed using g levels from the corresponding cask impact models described in Appendix 3D (both bottom-end drop and tipover) of the SAR. Analyses results indicate that the maximum normal and accident condition stresses are less than allowable values with a substantial margin of safety. Lid gasket compression is maintained at all times since bolt preload is higher than the applied loads during normal and accident condition loads.

3.3.4 Fuel Basket Analysis

Analysis of the fuel basket under various loading conditions is given in Appendix 3B of the SAR. Stress levels due to various loading is evaluated for the stainless steel boxes and plates, stainless steel fusion welds, and the aluminum side rails. Stresses are compared to allowable limits of ASME Section III, Division 1, Subsection NG (Core Support Structures). In a fashion consistent with other components, loads are classified as either normal or accident. A series of quasi-static analyses (using ANSYS) is used to estimate dynamic stresses under impact conditions. Impact loads for the basket resulting from tipover and end drop events are evaluated in Sections 3.3.1 and 3.3.2 above.

The TN-68 basket is analyzed for pressure and temperature loads during normal conditions. Temperature distributions in the basket are taken from Section 4 of the SAR. The effects of axial and radial thermal expansion of the basket are analyzed in Section 3B.3.4 of the SAR and found to have sufficient clearance for expansion. In addition, for conservatism, 3g vertical load and 1g side load are also applied to the basket as a bounding load for "normal conditions". The inertial loads of the fuel assemblies are applied as equivalent densities on the stainless steel boxes. A value of 705 lb. is assumed for the weight of each fuel assembly. Stress results for the "normal conditions" are presented in Section 3B.6.1 of the SAR. Large margins of safety are observed when compared to NG allowable levels. All stress values are less than 0.6 ksi, which is a small fraction of allowed stress. For this reason, the staff concludes that basket stress levels during normal operation are acceptable.

Response of the fuel basket for the tipover condition is discussed in Sections 3B.4.2 through 3B.4.4 and Sections 3B.5.1 through 3B.5.3 of the SAR. The former sections reference results from elastic analysis, while the latter sections reference results based upon plastic collapse loads. The two methods represent qualification based upon provisions of ASME Section III, Division 1, Appendix F, Articles F-1330 (elastic analysis) and F-1340 (plastic analysis) respectively. For both types of analysis, the basket "g" level assumed to occur during tipover is 77g's and is taken from Appendix 3D of the SAR.

Allowable stress values (Level D service limits) for the elastic analysis representing the various basket components are listed on page 3B.4-3 of the SAR along with the corresponding ASME Section III Appendix F references. A two-dimensional model described in Section 3B.3.2 of the SAR is used for the elastic evaluation of the basket and included the basket rails. Results are scaled from 1g results. The results representing 77g tipover are summarized in Tables 3.B.4-1 through 3.B.4-4 of the SAR. These tables represent results from three orientations of side drop (0, 30, and 45 degrees) and are structured to indicate peak "g" levels to obtain code allowables. The lowest such value is seen to be 94g's (well in excess of 77g's) and correspond to the stress limit of a fusion weld for the 45-degree impact orientation.

Elastic plastic analyses are performed by using a shell model of stainless boxes and stainless plates. The model is quasi-static and utilized large displacement effects. This model is shown in Figures 3B.5-2 and 3B.5-3 of the SAR. The ASME Article F-1340 acceptance criteria used is that of limiting loads to 70% of plastic instability load. Quasi-static loading corresponding to 200g's is initially applied for five radial orientations of the model (see Figure 3B.5-1 of the SAR). The most severely loaded model location is at the outer radius of the basket (location 1 of Figure 3B.5-2 of the SAR). While loads are greatest at this location, temperatures are not as great as for locations near the center of the basket. Since higher temperatures reduce the yield stress of the basket material, a second location (location 2 of SAR Figure 3B.5-2) is also investigated. Temperatures corresponding to 400 and 500 degrees F, respectively, were used at locations 1 and 2 for evaluating material properties.

Results for the plastic model are presented on pages 3B.5-3 and 3B.5-4 of the SAR. G levels representing 70% of the plastic instability g level are given. The values ranged from 101g's for location 1 and 158g's for location 2. Both are well in excess of the 77g's allowable value.

The plastic model described above does not involve any of the basket rail structure. The basket rails are evaluated separately in Section 3B.5.4 of the SAR. This is done by using

Subsection NF (Component Supports) and Appendix F of Section III, Division 1. The ligaments of the basket are analyzed as beam supports with combined axial compression and bending. It is demonstrated that the rails would qualify (using Level D service limits representing the hypothetical conditions) for 150g.

The staff has reviewed the "tipover" basket response analyses presented in the SAR. The following conservatisms are noted:

1. The equivalent side load is assumed to be 77g which will only occur at the upper end of the basket. In addition, this value reflects a dynamic amplification factor of 1.32. In reality, any basket compliance will tend to reduce this significantly.

2. Conservative (slightly higher than actual) temperatures are used in determination of yield strengths of the basket materials, thus making yield strengths less than actual.

3. The end conditions for "buckling type" calculations (the "K" values representing end fixity) is always assumed to represent pinned conditions. In reality, some degree of flexural stiffness will exist and tend to make the actual component less likely to buckle.

In view of these conservative assumptions and the substantial safety margins shown to exist, the staff believes that the basket has been adequately analyzed for "tipover" and that the basket would survive the tipover event.

Response of the fuel basket under 60g bottom-end drop is discussed in Section 3B.4.1 of the SAR. Quasi-static analyses showed stresses in the 304 assembly boxes to be about 6 ksi (42ksi allowable) while stress levels in the fusion welds in the boxes was shown to be 0.92ksi (21ksi allowable). Due to these large margins of safety and in view of the conservatism of the 60g assumption, the staff concludes that the basket is sufficiently designed to withstand an 18-inch bottom-end drop.

Based on the basket analysis, the staff concludes that the basket and rails will remain in place and maintain separation of adjacent fuel assemblies during both normal and accident conditions.

3.3.5 Spent Fuel Response due to Cask Impact Events

Appendix 6A of the SAR assesses the response of a typical BWR fuel assembly during end and tipover/side impact events. The analyses are quasi-static and utilize beam models to assess both axial and flexure response. Axial and transverse impact loading is assessed for a total of seven GE BWR fuel designs. The primary objective of the fuel response modeling is to assess the likelihood of gross fuel failure during such an event.

The methodology used in performing the fuel rod side impact stresses is based on work done at LLNL[13]. The fuel gas internal pressure is assumed to be present and the resulting axial tensile stress is added to the bending tensile stress due to 77g loads, which is taken from Section 3D.6.2 of the SAR (evaluated in Section 3.3.1 above). The stresses for different GE fuel assemblies as a result of the side drop accident are provided in SAR Table 6A-1 and are less than the yield stress of the irradiated zircaloy. Thus, the integrity of the fuel rods will not be breached during the tipover/side drop accident.

For the end drop study, large displacement finite element analyses using the ANSYS Finite Element Program are employed. The analyses use a three-dimensional finite element model of the entire active fuel rod length. Intermediate transverse supports are placed along the fuel to simulate grid support effects. The analyses also use the irradiated material properties and include the weight of fuel pellets. Internal pressure is neglected in the end drop study, since it produces tensile stresses in the cladding which in turn will reduce the compressive stresses caused by the end drop impact. Results from these analyses indicate that the lowest buckling load for GE fuel assembles is about 95g which is well above 60g for the cask bottom-end drop (evaluated in Section 3.3.2 above). Due to the conservatism inherent in the 60g end drop loads (described previously) and due to the large margins of safety relative to 60g's loading, the staff concludes that the fuel cladding tubes will not be damaged during a bottom-end drop accident.

The staff is in agreement with the conclusions of SAR Appendix 6A which indicate that gross failure of the fuel is unlikely for the anticipated impact events.

3.4 Extreme Natural Phenomena Events

3.4.1 Flood Condition

The TN-68 cask is analyzed for flood condition in Section 2.2.2 of the SAR. The cask is designed for an external pressure of 25 psi which is equivalent to a static head of water of approximately 57 feet. Based on a drag coefficient of 0.7 and a coefficient of friction of 0.2625, the analysis also indicates that water velocity of approximately 22 ft/sec along the entire height of the cask would be required to move the cask. Furthermore, the metallic seals in the cask are designed to maintain helium inside the confinement. They are also effective in preventing water in-leakage into the cask. Consequently, the staff concludes that the TN-68 cask will not be adversely affected by the flood condition.

3.4.2 Seismic Events

The TN-68 cask is analyzed for seismic loads in Section 2.2.3 of the SAR. The cask is conservatively considered as a rigid body placed on the concrete pad and equivalent static analysis methods are used to calculate loads and overturning moments. The coefficient of static friction of 0.35 and a lower bound cask weight of 218,000 lbs. are used to calculate the maximum frictional force available to prevent sliding. Based on the analyses, the TN-68 cask will neither slide nor tipover due to a seismic event with an applied horizontal acceleration of 0.26g and vertical acceleration of 0.17g. Because the minimum static coefficient of friction between the steel cask and the concrete pad could be as low as 0.3 in the references cited by the applicant, the TN-68 cask users will be required to verify that the coefficient of friction for their concrete pads is either greater than or equal to 0.35 as described in TS 5.2.1.

3.4.3 Tornado and Wind Loading

Loadings due to tornado and wind are addressed in Section 2.2.1 of the SAR. NRC Regulatory Guide 1.76[14] requires that the cask withstand the forces corresponding to a 360-mph tornado wind and a 3 psi pressure differential. Non-tornado wind loading is not significant in comparison to that due to tornados; therefore, the wind loading is bounded by the tornado loading. The analysis demonstrates that the cask will not tip, slide, or be otherwise damaged by this wind velocity loading or pressure loading. Lack of sliding is based on an assumed coefficient of friction of 0.35, which will be verified by the users of the TN-68 cask as discussed in Section 3.4.2 above.

3.4.4 Tornado Missile Impact

The analysis to determine the cask response to a tornado generated missile impact is provided in Section 2.2.1.2.2 of the SAR. The TN-68 cask stability is analyzed for four types of tornado missile impacts of 126 mph velocity, namely, Missile A - a 4,000 lb. automobile; Missile B - a 276 lb., 8-inch diameter projectile; Missile C - 1-inch diameter steel sphere; and Missile D- 4-inch thick wood plank 12-inch wide x 12-feet long was also analyzed with an assumed impact velocity of 300 mph. Based on the analyses, Missile A has the greatest effect on the stability of the TN-68 cask. It has the largest mass and produces the highest cask velocity after impact. The sliding analysis indicates that the TN-68 cask may slide 7.3 inches if Missile A strikes it below the cask center of gravity (CG). This sliding distance is calculated using the coefficient of dynamic friction of 0.2625. The coefficient of dynamic friction is approximately 25% smaller than the coefficient of static friction and is used when the cask begins to slide. Since the calculated sliding distance of 7.3 inches is much less than the distance between the two casks (approximately 94 inches), this would not cause the two casks to collide. The analyses further indicate that the TN-68 cask will not tipover due to Missile A striking above the cask CG, nor will there be any damage to the cask body. However, there could be localized damage to the neutron shield, protective cover, or overpressure monitoring system. Missiles B, C, and D may partially penetrate the cask wall if the energy is not first dissipated by the outer shell and neutron shield, but will not cause tipover. While the protective cover may be penetrated, it is shown that the lid would not be penetrated. Thus, the maximum damage would be limited to operational loss of the overpressure system.

3.4.5 Snow and Ice Loading

This loading condition is addressed in Section 2.2.4 of the SAR. Because of the heat load of the cask contents, the temperature of the protective cover attached to the top of the cask above the lid will generally stay above freezing. The protective cover is a 0.25-in thick torispherical steel head which can withstand an external pressure of more than 13 psi. By comparison, a 50 psf (0.35 psi) snow or ice load corresponds to approximately 6 ft of snow or 1 ft of ice. This load is insignificant on the protective cover. Therefore, the staff concurs that snow and ice loading has little structural consequence on the TN-68 cask.

3.5 Evaluation Findings

F3.1 SSCs important to safety are described in the TN-68 SAR, Revision 5 in sufficient detail to enable an evaluation of their structural effectiveness and are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events.

F3.2 The TN-68 storage system is designed to allow ready retrieval of spent nuclear fuel for further processing or disposal. The staff concludes that no accident or natural phenomena events analyzed will result in damage of the system that will prevent retrieval of the stored spent nuclear fuel.

F3.3 The cask is designed and fabricated so that the spent nuclear fuel is maintained in a subcritical condition under credible conditions. The configuration of the stored spent fuel is unchanged. Additional criticality evaluations are discussed in Section 6 of this SER.

F3.4 The cask and its systems important to safety are evaluated to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.5 The staff concludes that the structural design of the TN-68 dry storage cask is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the TN-68 cask system will enable safe storage of spent nuclear fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted practices, and confirmatory analysis.

3.6 References

1. TN-68 Dry Cask Safety Analysis Report, Rev. 5, Transnuclear Inc., May 1999.

2. NUREG-1536, Standard Review Plan for Dry Cask Storage System, January 1997.

3. ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1995 including 1996 addenda.

4. ANSI N14.6-1993, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More, June 1993.

5. WRC-107, Local Stresses in Spherical and Cylindrical Shells Due to External Loadings, 1965.

6. ANSYS Engineering Analysis System, User's Manual for ANSYS Rev. 5.2, 1995.

7. NUREG/CR-6608, Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet onto Concrete Pads, LLNL, February 1998.

8. LS-DYNA3D User's Manual (Nonlinear Dynamic Analysis of Structures in Three Dimensions), August 1995, Livermore Software Technology Corporation.

9. EPRI NP-4830, The Effect of Target Hardness on the Structural Design of Concrete Storage Pads for Spent-Fuel Casks, October 1986.

10. EPRI NP-7551, Structural Design of Concrete Storage Pads for Spent-Fuel Cask, August 1991.

11. Rashid, Y.R., R. J. James and O. Ozer, Validation of EPRI Methodology for Analysis of Cask Drop and Tipover Accidents at Spent Fuel Storage Facilities, May 1997.

12. NUREG/CR-6007, Stress Analysis of Closure Bolts for Shipping Casks, LLNL, April 1992.

13. LLNL Report UCID-21246, "Dynamic Impact Effects on Spent Fuel Assemblies," October 1987.

14. U.S. Nuclear Regulatory Commission Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants, April 1974.

4.0 THERMAL EVALUATION

The thermal review of Revision 5 of the TN-68 SAR ensures that the cask and fuel material temperatures of the TN-68 Dry Storage Cask system will remain within the allowable values or criteria for normal, off-normal, and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation which could lead to gross rupture. This review also confirms that the thermal design of the cask has been evaluated using acceptable analytical and/or testing methods.

4.1 Spent Fuel Cladding

The staff verified that the analyzed cladding temperatures for each fuel type proposed for storage are below temperatures which could cause cladding damage that would lead to gross rupture. For normal conditions of storage, the applicant calculated a limiting BWR fuel cladding temperature of 649°F (343°C). This limit is based on internal fuel rod pressure according to the methodology of PNL-6189[1] and is acceptable to the staff. For the short-term accident and loading/unloading operations, the applicant used the temperature limit of 1058°F (570°C) from PNL-4835[2]. This limit is acceptable to the staff for short-term conditions.

In Section 4.6 of the SAR, the applicant considered the effect of cladding integrity during cask reflood operations that quench the hot spent fuel. The applicant provided a quench analysis of the fuel in SAR Section 3.5.2 that concluded the total stress on the cladding as a result of the quenching process is below the cladding materials minimum yield strength.

4.2 Cask System Thermal Design

4.2.1 Design Criteria

The design criteria for the TN-68 storage cask have been formulated by the applicant to assure that public health and safety will be protected during dry cask spent fuel storage. These design criteria cover both the normal storage conditions for the 20-year approval period and postulated accidents that last a short time, such as a fire.

Section 4.1 of the SAR defines several primary thermal design criteria for the TN-68 cask:

1. The allowable seal temperatures must be within the specified limit of 536°F (280 °C) to satisfy the leak tight confinement function during normal storage conditions.

2. Maintenance of the neutron shield resin during normal storage conditions; an allowable range of -40 to 300°F (-40 to 149°C) is set for the neutron shield.

3. Maximum and minimum temperatures of the confinement structural components must not adversely affect the confinement function.

4. The short-term allowable cladding temperatures that are applicable to off-normal and accident conditions of storage are based on PNL-4835.

5. The allowable fuel cladding temperatures to prevent cladding degradation during long-term dry storage conditions are provided in Section 3.5.1 of the SAR.

The staff concludes that the primary thermal design criteria have been sufficiently defined.

4.2.2 Design Features

To provide adequate heat removal capability, the applicant designed the TN-68 system with the following features:

1. Helium backfill gas for heat conduction which also provides an inert atmosphere to prevent fuel cladding oxidation and degradation;

2. Minimal heat transfer resistance through the basket by sandwiching aluminum neutron poison plates between the stainless steel fuel compartments. The compartments are fusion welded to 1.75 in. wide stainless steel plates. Above and below the plates are slotted poison plates, which form an egg crate structure providing good paths for heat transfer from the fuel assemblies, along the plates, to the aluminum basket rails.

3. The basket rails are bolted to the basket periphery providing a good conduction path to the cask cavity wall.

4. Aluminum boxes filled with a resin compound are placed around the cask gamma shell and enclosed by an outer shell. The boxes provide for neutron shielding and increase the thermal conductance through the neutron shield layer.

5. High emissivity paint on the exterior cask surface to maximize radiative heat transfer to the environment.

The staff verified that all methods of heat transfer internal and external to the TN-68 are passive. Drawings in Section 1.5 of the SAR along with the summary of material properties in SAR Section 4.2, Tables 1 to 9, provide sufficient detail for the staff to perform an in-depth evaluation of the thermal performance of the entire package as required by 10 CFR 72.24(c)(3)[3].

4.3 Thermal Load Specifications

The design-basis fuel to be stored in the TN-68 cask is described in Section 2.1 and Tables 2.1-1, 2.1-2, and 2.1-3 of the SAR for the BWR fuel. The TN-68 cask is designed to dissipate 21.2 kW or 0.312 kW/assembly. The axial profiles for the design-basis fuels are in SAR Section 4.4.1. The peak power in the BWR assemblies is a factor of 1.2 times the average power. Maximum fuel assembly heat load is given in TS 2.1. By review and confirmation using independent analysis, the staff has reasonable assurance design-basis decay heats were determined properly.

4.3.1 Storage Conditions

To bound the normal storage, off normal, and design-basis natural phenomena conditions, the applicant defined two external environments for storage conditions in Sections 4.1 and 4.4 of the SAR. The maximum storage condition considers a 100°F (37.8°C) average daily temperature and includes solar insolation equivalent to the total 10 CFR 71.71(c)[4] insolation averaged over a 24-hour period. The total 10 CFR 71.71(c) insolation in a 12-hour period is 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. Since heat transfer through the top and bottom of the cask was ignored, only insolation from the curved side surfaces was considered. The minimum storage condition considers a –20°F (-28.9°C) average daily temperature and assumes no solar insolation. Each external environment included analyses for heat loads corresponding to 0, 20, and 40 year storage as shown in SAR Tables 4.4-2 and 4.4-3. The staff concludes that the applicants approach of using maximum and minimum daily average temperatures and insolation for the TN-68 cask is acceptable because cask temperature response to changes in the ambient conditions will be slow due to the large thermal inertia of the cask. Maximum and minimum average daily temperatures are included in TS 4.3 as siting parameters that must be evaluated by the cask user.

4.3.2 Accident Conditions - Fire

The fire accident postulated for the TN-68 storage cask is described in Section 4.5.1 of the SAR. The cask initial temperature distribution before the postulated accident is based on the zero year maximum storage conditions.

A 15 minute fire with an average flame temperature of 1550°F(843°C), an average convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F, and an emissivity of 0.9 are hypothesized. This is postulated to be caused by the spillage and ignition of 200 gallons of combustible transporter fuel. The assumed 15-minute duration for the transient evaluation is based on a calculated fire duration of 13 minutes for this amount of fuel. Staff calculations of the fire duration agreed with the applicant.

Following the fire, the outside environment is restored to the maximum storage conditions and the TN-68 cask transient analysis is continued to evaluate temperature peaking of cask components. Based on review, the staff concludes that the thermal loads for the fire accident are acceptable.

4.3.3 Accident Conditions - Buried Cask

The buried cask accident postulated for the TN-68 is described in Section 4.5.2 of the SAR. The cask initial temperature distribution before the postulated accident is based on the 0 year maximum storage conditions. The TN-68 cask normally dissipates heat to the environment via radiation and convection. For this accident, the applicant assumed the burial media effectively insulated the cask outer surfaces. The analysis then determines the time to reach limiting temperatures for confinement integrity. Based on review, the staff concludes that the thermal loads for the cask burial accident are acceptable.

4.3.4 Cask Heatup During Loading

For cask heatup during vacuum drying, the cask has been drained of water and filled with air. Initial cask temperatures of 115°F (46°C) , a building ambient temperature of 115°F(46°C), and a maximum allowable cask heat load of 21.2 kW were assumed. The heatup analysis assumed only conduction through air and neglected convection and radiation between the basket and the cask wall. Based on review, the staff concludes that the thermal loads for cask heatup are acceptable.

4.4 Model Specification

4.4.1 Configuration

A three-dimensional model for thermal design of the TN-68 system was developed using the finite element ANSYS[5] computer code. Transport of heat from the fuel assemblies to the outside environment is analyzed using a single large model of the TN-68 cask standing vertical on the concrete pad. The fuel region is modeled as a homogenized material with an effective thermal conductivity for the fuel. All other cask components are modeled in detail. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and thermal radiation heat transfer mechanisms from the cask vertical surfaces.

The staff reviewed the applicants use of the ANSYS computer code and the associated inputs, assumptions, material properties, boundary conditions, and initial conditions. The staff has reasonable assurance that the temperatures of the cask components and the cask pressures under normal and accident conditions were determined correctly. Details of the modeling assumptions and approach follow.

4.4.1.1 Fuel Assembly Model

Heat transfer through the fuel assemblies was modeled by treating the fuel region as an homogenized material with effective thermal conductivity's (K_{eff}) determined for the transverse and axial directions. First, the applicant used the modified Wooton-Epstein correlation to calculate the K_{eff} of the various fuel assemblies designated for the TN-68 and determine the bounding BWR fuel type. The GE12 10x10 BWR fuel assembly yielded the highest cladding temperatures and was therefore selected as the bounding assembly for detailed analysis to define the fuel K_{eff}.

For the GE12 10x10 BWR fuel assembly, the axial effective thermal conductivity was calculated based on the parallel paths of heat conduction through the cladding and the helium fill gas. Axial conduction through the fuel pellet was neglected. The transverse effective thermal conductivity was determined by using the ANSYS computer code to model a detailed two-dimensional quarter symmetry section of the GE12 10x10 fuel assembly. A series of simulations with varying temperature boundary conditions was performed. The temperature drop across the assembly was then related to the K_{eff} of the fuel. A resultant relationship of K_{eff} of the fuel verses average temperature of the assembly was developed. The effective specific heat and density for the homogenized fuel assemblies were determined using a mass weighted average approach.

4.4.1.2 TN-68 Basket Section Model

The heat rejection capability of the TN-68 design was evaluated by developing a thermal model of the homogenized fuel assemblies, the basket wall geometry, and the layers that form the cask body. The ANSYS model includes the geometry and materials of the basket, the basket rails (peripheral inserts), the cask shells, the neutron shielding (resin in the aluminum containers), and the outer shell.

A detailed quarter slice section, the length of the neutron shield, of the TN-68 cask was modeled with the appropriate symmetry boundary conditions. The model is shown in SAR Figures 4.4-2, 4.4-3 and 4.4-5. The decay heat from the fuel assemblies was applied to the homogenized fuel elements as volumetric heat generation in the 144-inch active fuel length.

The model includes 17 of the 68 basket stainless steel boxes joined by fusion welding to 1.75-inch stainless steel plates. Slotted poison plates (0.30-inch thick) that form an egg crate structure are located above and below the stainless steel plates. The thermal model accounts for heat transfer through the stainless steel plates, the aluminum poison plates and the stainless steel boxes. Aluminum basket rails, bolted to the basket periphery, increase the surface area for heat dissipation, while providing structural support for the basket.

Adjacent basket structural components were assumed to have the following gaps:

- 0.01-inch gap between the aluminum poison plate/stainless steel plates and the stainless steel fuel compartments.
- 0.125-inch gap between the rails and the basket periphery.
- 0.125-inch gap between the basket rails and the cask cavity walls.
- 0.06-inch gaps between plates in the axial direction.

Only conduction through the helium gas is modeled across gaps. Radiative and convective heat transfer were neglected.

4.4.1.3 Cask Body Model

From the inner cavity wall to the exterior cask surface, heat is conducted though an array of concentric layers representing the confinement shell, the gamma shield shell, the resin filled boxes that form the neutron shield shell, and the outer shell. Heat rejection from the cask exterior surfaces to ambient air includes natural convection and thermal radiation heat transfer from the vertical surfaces.

4.4.1.4 Radiation from Cask Exterior Surfaces

The applicant considered the thermal radiation interaction among casks in an array. The radiation from the cask was lowered to account for an array of casks. A two cask wide and infinitely long array was assumed. This assumption resulted in an overall view factor of 0.70 between the cask vertical surfaces and the ambient environment. To ensure the assumed radiative configuration is maintained during storage, a minimum center-to-center cask spacing of 16 feet is included in TS 4.2.1.

4.4.2 Material Properties

The material properties used in the thermal analysis of the storage cask system are listed in SAR Section 4.2, Tables 1 to 9. The applicant provided a summary of the material compositions and thermal properties for all components used in the cask model. The material properties given reflect the accepted values of the thermal properties of the materials specified for the construction of the cask. For homogenized materials such as the basket walls, the applicant described the manner in which the effective thermal properties were calculated.

4.4.3 Boundary Conditions

The boundary conditions include the total decay heat and the external conditions on the cask surface. The axial peak power for the BWR assemblies is a factor of 1.2 times the average power of 0.312 kW/assembly. The cask external boundary conditions depend on the environment surrounding the cask and are detailed below.

4.4.3.1 Storage Conditions

For storage conditions, the applicant included boundary conditions for ambient temperatures and insolation as described in SER Section 4.3.1. Heat transfer from the top and bottom of the cask were neglected in favor of conservatism.

4.4.3.2 Accident Conditions - Fire

For the postulated fire accident conditions, the finite element model described in SAR Section 4.4.1 for the storage condition was modified by reducing the geometry into a 3-dimensional slice model located at hottest axial location in the cask. In addition, a lid seal region was modeled to examine temperatures at the cask seals. These models are shown in Figures 4.5-1 and 4.5-2 of the SAR. The boundary conditions include the cask initial temperature distribution before the postulated accident, fire conditions for the fire transient, and ambient temperatures and insolation post fire as described in SER Section 4.3.2.

4.4.3.3 Accident Conditions Buried Cask

For the postulated buried cask accident conditions, the same finite element model described in SAR Section 4.5.1.1 for the fire accident was applied. The boundary conditions include the cask initial temperature distribution before the postulated accident and an essentially insulated condition for the transient as described in SER Section 4.3.3.

4.4.3.4 Cask Heatup Analysis

For the cask heatup analysis, the finite element model described in SAR Section 4.5.1.1 for the fire accident was modified slightly. In this case all gaseous heat conduction within the cask cavity is through air instead of helium. Radiation heat transfer between the basket and the cask wall was neglected. The boundary conditions include the initial cask and ambient temperatures and the cask environment during the transient as described in SER Section 4.3.4.

4.5 Thermal Analysis

4.5.1 Computer Programs

The thermal analysis was performed using the ANSYS finite element modeling package. ANSYS is capable of general 3-D steady-state and transient calculations. The output from the code work was plotted in SAR Figures 4.4-7, 4.4-8, 4.5-3, 4.5-4, and 4.6-1 and discussed in SAR Section 4.4.

4.5.2 Temperature Calculations

4.5.2.1 Storage Conditions

The TN-68 cask has been analyzed to determine the temperature distribution under long-term storage conditions that envelop normal, off-normal, and design basis natural phenomena conditions. The basket is considered to be loaded at design-basis maximum heat loads with BWR assemblies. The systems are considered to be arranged in an ISFSI array and subjected to design-basis ambient conditions with insolation. The maximum allowable temperatures of the components important to safety are discussed in Section 4.1 of the SAR. Low temperature conditions were also considered.

The calculated fuel clad temperatures for zircaloy-clad fuel assemblies are listed in SAR Table 4.4-1. Temperature criteria for the spent fuel cladding are discussed in Section 4.2 of the SER. The fuel cladding temperatures remain below their acceptable temperatures. SER Table 4-1, below, summarizes the temperatures of key components in the cask for various environmental conditions.

Table 4-1 Temperatures of Key Components							
Component	Storage Conditions			Fire Accident		Burial Accident	
	Maximum (°F)	Minimum* (°F)	Allowable Range (°F)	Peak (°F)	Allowable Range (°F)	Time to limit (hours)	Allowable Range (°F)
Outer Shell	218	-20	**	847	**	**	**
Lid	***	-20	**	287	**	**	**
Seal	247	-20	-40 to 536	340	-40 to 536	64	-40 to 536
Radial Neutron Shield	259	-20	-40 to 300	N/A	N/A	12	N/A
Inner Shell	277	-20	**	346	**	**	**
Cask Bottom	269	-20	**	***	**	**	**
Ave Gas Cavity	379	-20	N/A	416	**	**	N/A
Basket Rail	330	-20	**	377	**	**	**
Basket Plate	482	-20	**	532	**	**	**
Fuel Cladding	502	-20	649 max.	550	1058 max.	177	1058 max.

* Assuming no credit for decay heat and a daily average ambient temperature of -20°F

** The components perform their intended safety function within the operating range.

*** Not modeled.

4.5.2.2 Accident Conditions - Fire

The applicant analyzed a fire accident on the TN-68 cask using the conditions specified earlier in Section 4.4.3.2 of the SER. The peak temperatures of the key cask components due to a 15-minute fire with a 21.2 kW decay heat are shown in SER Table 4-1. The initial temperatures are based on the maximum storage conditions. All of the fire accident temperatures were below the short-term design-basis temperatures with the exception of the neutron shield material. However, as discussed in SER Sections 5 and 11, the accident condition dose rate limits are shown to remain below the regulatory limit of a total dose of 5 rem assuming complete removal of the neutron shield. Based on these analyses, the staff has reasonable assurance that the cladding integrity and the confinement boundary will not be compromised during the fire or post-fire transient and doses with a damaged neutron shield will remain within limits.

4.5.2.3 Accident Conditions Buried Cask

The results for this accident are summarized in SER Table 4-1. The neutron shield temperature limit is reached at 12 hours, the seal limit at 64 hours and the cladding limit at 177 hours. Based on review, the staff concludes that the thermal analysis of an adiabatic heatup resulting from cask burial is acceptable. As discussed in SAR Section 11.2.10, corrective actions to un-bury the cask will need to be taken as soon as possible to protect the seal and cladding integrity.

4.5.2.4 Cask Heatup Analysis

The applicant performed a transient analysis of the cask heatup prior to being filled with helium using the conditions specified in Section 4.4.3.4 of the SER. Assuming design basis heat load fuel, vacuum drying and helium backfill must be completed within 48 hours to maintain cask component temperatures below their allowable temperatures. At 48 hours, the maximum fuel cladding temperature of 616°F (324°C) was predicted which is well below the fuel cladding short term temperature limit of 1058°F.

TS 3.1.1 and 3.1.2 are constructed to ensure that the vacuum drying and helium backfill operations are completed within 48 hours or corrective action such as injecting a partial pressure of helium in the cask is required. The applicant concluded that a helium partial pressure of 0.1 atm (the remaining 0.9 atm is assumed to be air) would sufficiently improve heat transfer to maintain temperatures below limits. The staff calculated the effective gas mixture conductivity of the air-helium mixture and determined that the net conductivity was about 25% that of pure helium. Data from Interim Staff Guidance (ISG) 7[6], documents the sensitivity of a different storage cask system to a net gas mixture conductivity of 30% that of pure helium. That data show that fuel cladding and bulk gas temperatures would increase about 3%. Post fire transient temperatures of cask cavity components are about 8% greater than the steady state temperatures. Based on the above, the staff concludes that there is reasonable assurance that the cask component and fuel cladding temperatures will be maintained within limits if a 0.1 atm partial pressure of helium is maintained in the cask for short term operations.

4.5.3 Pressure Analysis

4.5.3.1 Storage/Off Normal/Accident Conditions

In SAR Sections 7.2.2 and 7.3.2.2, the applicant evaluated internal pressurization for the following conditions:

1) 100°F (37.8°C) ambient air temperature and insolation (maximum storage conditions)

2) maximum storage conditions and 10% fuel rod failure (off-normal)

3) maximum storage conditions and 100% fuel rod failure (accident)

4) maximum storage conditions, 100% fuel rod failure, and 15 minute external fire (fire accident)

5) 100°F (37.8°C) ambient air temperature, 100% fuel rod failure, and cask burial under debris (burial accident)

The staff reviewed the applicants calculations and performed confirmatory analyses. The applicants calculations used appropriate methods and cover gas temperatures determined in SAR Section 4. The highest predicted pressure was 63 psig at a cavity gas temperature of 620°F (327°C) for the cask burial accident. Staff calculations were in agreement with the applicants results and confirmed that the expected pressures were below the design internal pressure of 100 psig. Based on review and confirmatory analyses, the staff concluded that internal cask pressures remain below the cask design pressure rating under normal, off-normal, design-basis natural phenomena, and design-basis accident conditions or events.

4.5.3.2 Pressure during Unloading of Cask

In SAR Section 4.6.1, the applicant considered the transient resulting from reflooding the cask with water prior to placement of the cask in the spent fuel pool for fuel unloading. To control cask pressurization, the maximum initial cask reflood rate is controlled to maintain internal pressure below 90 psia (design pressure 100 psig). A TN calculation[7] showed that the maximum saturated steam flow rate from the cask was 0.144 lbm/sec at 90 psia assuming the cask vent discharged into the spent fuel pool as shown in SAR Figure 8.2-1. To limit the water supply for steam production to less than the predicted maximum discharge mass flow rate, operating procedure descriptions in SAR Section 8.2 reflect a maximum initial inlet flow rate of 0.140 lbm/sec (1.0 gallon per minute). The applicant further ensures over-pressure protection of the cask during reflood by placing a valve in the water inlet line that will restrict cooling water flow if inlet pressure reaches 90 psia. Based on staff review of the reflood evaluation and the proposed operational controls, the staff has reasonable assurance that the cask can be maintained within design pressure limits during reflood.

4.5.3.3 Pressure during Loading of Cask

Although the cask is vented during the draining procedure, the applicant evaluated cask pressurization during loading operations when the cask is filled with water and removed from the spent fuel pool. A bounding case, where the maximum allowable cask heat load of 21.2 kW is applied to the boiling of the cask water, was analyzed to determine if the cask design pressure of 100 psig would be reached. The applicant determined that a rate of evaporation of 0.023 lbm/sec is much less than the flow rate of 0.144 lbm/sec that was calculated for a cask pressure of 90 psia. The staff performed a confirmatory calculation of the evaporation rate, found it to be in agreement with the applicant, and therefore concluded that there is reasonable assurance that the cask pressure will remain below design pressure during loading operations.

4.5.4 Confirmatory Analyses

The confirmatory analyses of the TN-68 storage cask SAR can be divided into six categories: (1) review of models used in the analyses, (2) review of material properties used in the analyses, (3) review of boundary conditions and assumptions, (4) perform independent, confirmatory analyses, (5) compare the results of the analyses with the applicant's design criteria, and (6) assure that the applicant's design criteria will satisfy the regulatory acceptance criteria and regulatory requirements.

The staff reviewed the approaches used by the applicant in the thermal analyses. The staff performed a confirmatory analysis of the thermal performance of the cask systems, structures, and components identified as important to safety. A detailed model of the TN-68 Cask was developed using the COBRA-SFS[8] computer code to evaluate the SAR results. The temperature distributions generated by the staff's model displayed good agreement with those values determined by the applicant.

The staff performed a sensitivity study on the impact of gap sizes in the cask body layers on the thermal performance of the design. The applicant assumed 0.01-inch gaps between the aluminum neutron shield boxes and the adjacent gamma shield shell and surrounding outer shell. Using a linear scale up, the staff projected a peak cladding temperature lower than the long term storage cladding temperature limit of 649°F (343°C) if fabrication results in gaps of 0.05 -inches or smaller between the component layers.

The staff has determined that the thermal SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness. Based on the applicants analyses, there is reasonable assurance that the TN-68 cask is designed with a heat removal capability having testability and reliability consistent with its importance to safety. The staff further concludes, based on review and confirmatory analysis, that there is reasonable assurance that analysis of the TN-68 cask demonstrates that the applicable design and acceptance criteria have been satisfied.

4.6 Evaluation Findings

F4.1 SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness.

F4.2 The staff has reasonable assurance that the decay heat loads were determined appropriately and accurately reflect the burnup, cooling times, and initial enrichments specified.

F4.3 The staff has reasonable assurance that the temperatures of the cask SSCs important to safety will remain within their operating temperature ranges and that cask pressures under normal and accident conditions were determined correctly.

F4.4 The staff has reasonable assurance that the TN-68 cask is designed with a heat removal capability having testability and reliability consistent with its importance to safety.

F4.5 The staff has reasonable assurance that the TN-68 cask provides adequate heat removal capacity without active cooling systems.

F4.6 The staff has reasonable assurance that the spent fuel cladding will be protected against degradation that leads to gross ruptures by maintaining the clad temperature below maximum allowable limits and by providing an inert environment in the cask cavity.

F4.7 The staff concludes that the thermal design of the TN-68 is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the TN-68 will allow safe storage of spent fuel for a certified life of 20 years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

4.7 References

1. Levy, I. S., et al., Pacific Northwest Laboratories, “Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircaloy Clad Fuel Rods in Inert Gas,” PNL-6189, May 1987.

2. Johnson, A. B. and E.R. Gilbert, Pacific Northwest Laboratories, “Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases,” PNL-4835, September 1983.

3. U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear and High-level Radioactive Waste,” Title 10, Part 72.

4. U.S. Code of Federal Regulations, “Packaging and Transportation of Radioactive Material,” Title 10 Part 71.

5. ANSYS Engineering Analysis System, “Users Manual for ANSYS, Revision 5.4,” ANSYS Inc., Houston, PA., September 1997.

6. Kane, W. F., "Spent Fuel Project Office Interim Staff Guidance 7," ISG-7. U. S. Nuclear Regulatory Commission, October 1998.

7. Transnuclear Inc., “Mass Flow Rates During Unloading,” Calculation No. 1066-70, Revision 0, January 1999.

8. Michener, T. E., et. al., Pacific Northwest Laboratories, COBRA-SFS: “A Thermal-Hydraulic Analysis Code fore Spent Fuel Storage and Transportation Casks,” PNL-10782, September 1985.

5.0 SHIELDING EVALUATION

The shielding review evaluates the capability of the TN-68 shielding features to provide adequate protection against direct radiation from its contents. This review considered dose rate calculations from both photon and neutron radiation at locations near the cask and at specific distances away from the cask. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d)[1,2]. An overall assessment of compliance with 10 CFR Part 72 dose limits for members of the public is discussed in Section 10 (Radiation Protection) of the SER and includes direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations.

5.1 Shielding Design Features and Design Criteria

5.1.1 Shielding Design Features

The TN-68 cask is designed to provide both photon and neutron shielding. The principal components of the radial photon shielding are the 1.5-inch thick steel inner shell, the 6.0-inch thick steel body wall, and a 0.75-inch thick steel outer shell. Photon shielding at the bottom of the cask is provided by the 1.5-inch thick steel base of the confinement vessel and an 8.25-inch thick steel bottom plate. The photon shielding at the top of the cask consists of a 4.5-inch thick steel shield plate and a 5.0-inch thick steel lid. In addition, there is a provision for an optional 1-inch thick steel shell above the radial neutron shield. However, as specified in Technical Specification 5.2.3, this shield is required when needed to meet the dose-rate limits in the specification.

In addition to the steel components discussed above, radial neutron shielding is provided by a borated polyester resin compound cast into long slender aluminum containers that surround the cask body. The thickness of the radial neutron shield material is 6.0-inches. The top neutron shield material is 4.0-inches of polypropylene.

5.1.2 Shielding Design Criteria

The overall design criteria for the TN-68 cask are the regulatory dose limits and requirements in 10 CFR Part 20, and 10 CFR 72.104(a), and 10 CFR 72.106(b).

The staff evaluated the TN-68 shielding design features and design criteria and found them to be acceptable. The SAR analysis indicates reasonable assurance that the shielding design features and design criteria can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Cask surface dose rates are to meet specific limits as described in TS 5.2.3.

An evaluation of the overall radiation protection design features and design criteria of the TN-68 cask is given in Section 10 of the SER.

5.2 Radiation Source Definition

5.2.1 Source Specification

The radiation source specification is presented in Section 5.2 of the SAR. Photon and neutron source terms were generated with the SAS2H and ORIGEN-S modules of SCALE 4.3, using the 45-group coupled neutron-photon cross-section library[3]. The source terms for the fuel assembly designs specified in SAR Table 2.1-1 were examined to determine which assembly gives the maximum source terms. The examination included the photon and neutron sources from the fuel, the photon sources from activated assembly hardware, and an estimate of the dose rates on the cask side at the surface and at 1 meter. The applicant determined that the GE 7x7 fuel assembly with 197.7 kg of uranium resulted in the maximum source terms and total dose rates. Consequently, design-basis source terms were calculated for Zircaloy clad fuel in the GE 7x7 fuel assembly.

In the hardware activation analysis, cobalt impurities in Inconel, Zircaloy, and stainless steel were assumed to be 6490, 10, and 800 ppm, respectively. The applicant obtained these values from Reference 4. The values are recommended by the Oak Ridge National Laboratory after a review of the available materials documents and specifications. Based on actual measurements as reported in Reference 5, the applicant's values for Inconel and Zircaloy bound the measured values, and the value for stainless steel, while not absolutely bounding, is an acceptable representation of typical hardware. Although the cobalt impurities in fuel assembly hardware can vary depending on the supplier and date of manufacture, the applicant's assumed values are reasonable and acceptable.

To correct for spatial and spectral changes of the neutron flux outside the fuel zone during irradiation, the masses of the materials in the bottom end fitting, plenum, and top end fitting were multiplied by scaling factors of 0.15, 0.2, and 0.1, respectively. These are the factors recommended in Reference 5. These scaling factors produce calculated source terms which bound measured source terms. The staff performed confirmatory calculations with ORIGEN2[6] and obtained Co-60 source terms which are 20% lower than the values in the SAR. The neutron flux scaling factors from Reference 5 are derived from measurements and are considered to provide bounding values.

The characteristics of the GE fuel assemblies are given in SAR Tables 5.2-1 and 5.2-1a. The design-basis 7x7-fuel assembly gives the largest source terms as shown in SAR Table 5.2-4. A minimum U-235 enrichment of 3.3 wt% at a burnup of 40,000 MWD/MTU and a cooling time of 10 years maximizes the radiation source terms. The source terms for this fuel are listed in SAR Tables 5.2-7, 5.2-9, and 5.2-10. The source terms for the fuel region include photon radiation from activated channels. The limits on burnup and cooling time for fuel of lower enrichments are provided in SAR Section 2 and are given in Table 2.1-4 of the SAR.

The staff performed confirmatory analyses of the design-basis photon and neutron source terms using the ORIGEN2 code. The staff has examined the Department of Energy (DOE) Characteristics Data Base[7] and determined that BWR fuel burned to 40,000 MWD/MTU could have an enrichment as low as 2.74 wt% which would increase the neutron source by 45%. To preclude fuel of this type with a 10-year cooling time from being loaded into the TN-68 cask, a TS in the form of Table 2.1.1-1 is included. Fuel of lower enrichment than 3.3% may be stored in the TN-68 cask provided the burnup is lower and/or the cooling time is increased according to TS Table 2.1.1-1. The staff has reasonable assurance that the design-basis photon source terms are adequate for the shielding analysis.

The applicant calculated a neutron source term assuming all of the fuel was irradiated to the design-basis exposure. Since the neutron source increases exponentially with burnup, neutron peaking factors based on an axial burnup profile were calculated and are listed in SAR Table 5.2-10. The integration of the neutron source as a function of axial position resulted in a 32% larger total neutron source than that given in Table 5.2-4 of the SAR for the average burnup value. It is staff's opinion that the water densities in the neutron source term calculation, used by the applicant toward the top of the fuel, are not bounding; thus, the neutron source is under-estimated in this region. Staff calculated a neutron source term that was 14% higher than the applicant's final value. However, the difference in neutron source is offset by the applicant's higher photon source term and does not change the conclusions about the cask meeting safety requirements.

The axial burnup profile is given in SAR Table 5.2-8 and shown graphically in Figure 5.2-1. The profile is representative of modern-day BWR fuel which has natural uranium blankets on the ends; 6 inches on the bottom and 12 inches on the top. As shown in Table 5.2-1, a large part of this fuel has an active length of 150 inches. Older BWR fuel does not have natural uranium blankets; consequently, the neutron and photon peaking factors at the ends of the fuel are higher than the values given in Table 5.2-10. Since older BWR fuel generally was discharged at burnups much lower than the design-basis burnup of 40,000 MWD/MTU and the decay time is longer than 10 years, the neutron source calculated by the applicant is expected to be bounding for most cask loadings. This conclusion is supported by the fact that the confirmatory calculations, which used the higher source terms, gave lower total dose rates than the applicant at the cask mid-plane and above the neutron shield.

The applicant's computational models divided the axial burnup profile into 6 burnup zones in the top half of the fuel and 6 burnup zones in the bottom half of the fuel with smaller zones at the ends of the fuel. The relative neutron and photon source distributions are given in SAR Table 5.2-10.

5.3 Shielding Model Specifications

5.3.1 Model Specifications

The model specifications for shielding are presented in Section 5.3 of the SAR. The applicant's shielding model for normal and accident conditions consists of a 3-D representation of the TN-68 cask using the design drawings in Section 1.5 of the SAR. A description of the shielding configuration is presented in Section 5.3.1 of the SAR. Radial views of the shielding model are depicted in SAR Figures 5.3-1 thru 5.3-5. Axial views of the shielding model are depicted in SAR Figures 5.3-1, 5.3-2, 5.3-4, and 5.3-5.

5.3.1.1 Source Region Configuration

The radiation source is divided into four major axial regions: bottom end fitting, fuel, plenum, and top end fitting. The relative positions of these source term regions are also depicted in the axial view figures identified above. The self shielding mass in the fuel region is modeled as a totally homogeneous material, and the end fittings and plenum regions are modeled as homogeneous regions of stainless steel, Inconel, and Zircaloy.

The axial distributions of the radiation source terms in the fuel region were developed from actual utility operator burnup data. The photon source profile is listed in SAR Table 5.2-10. The axial distribution of the neutron source was determined from a series of SAS2H/ORIGEN-S calculations using the axial burnup profile in SAR Table 5.2-8 and SAR Figure 5.2-1. The neutron source profile is listed in SAR Table 5.2-10. This profile was used to account for the non-linear buildup of neutron source terms (primarily Cm-244) as a function of burnup. The photon source distributions within the plenum, top end fittings, and bottom end fittings were assumed to be uniform.

5.3.1.2 Streaming Paths and Regional Densities

The applicant's shielding models included streaming paths for the trunnions. The cask design shields other potential streaming paths. Simplified model in the confirmatory calculations indicates that dose rates at the trunnions are greater than dose rates above and below the radial neutron shield and at the mid-plane. However, the model for the confirmatory calculations did not model the trunnions in detail and the area of concern is localized.

The composition and densities of the materials used in the shielding analysis are presented in SAR Table 5.3-1. The applicant did not identify any materials that undergo changes in material density or composition from temperature variations. The bounding accident condition for shielding assumes complete loss of both the radial neutron shield and the axial neutron shield.

The staff evaluated the SAR shielding models and found them to be acceptable. The basket and fuel inserts were homogenized with the fuel rods and assembly hardware for the radial calculations but not for the axial calculations. The model dimensions and material specifications are consistent with the drawings in Section 1 of the SAR, and thus, provides the basis for reasonable assurance that the TN-68 cask was adequately modeled in the shielding analysis.

The staff notes that the aluminum tubes containing the neutron shield material were homogenized with the neutron shield. Staff concludes that since the aluminum tubes have a wall thickness of only 1/8 -inch and actual measurements have not detected streaming, neutron streaming through the aluminum is insignificant.

5.4 Shielding Analyses

5.4.1 Shielding Analyses

The shielding analyses are presented in Section 5.4 of the SAR. The 3-D Monte Carlo transport code, SAS4[8], was used for the cask surface and 1 meter shielding analysis. The cross-section library used has 27 neutron groups and 18 photon groups and is based on ENDF/B-IV cross-section data. The Monte Carlo N-Particle (MCNP) code[9] was used to calculate direct and skyshine dose rates at long distances. The SAR uses the ANSI/ANS

Standard 6.1.1-1977 flux-to-dose-rate conversion factors to calculate dose rates in the shielding analysis.

5.4.1.1 Normal Conditions

The SAR presents calculations for normal condition dose rates from the design-basis fuel for the TN-68 cask. Calculated dose rates listed in SAR Table 5.1-2 are averages over the specified surface area or 1 meter from the surface. The higher dose rates on the side of the cask occur above and below the radial neutron shield. The average values for these regions are 397 mrem/hr and 267 mrem/hr, respectively. The dose rates above the radial neutron shield assume that the optional shield ring is in use.

The staff performed confirmatory calculations for the TN-68 cask using the MCNP code. The MCNP model included the optional shield ring. The MCNP model is similar to the SAS4 model used by the applicant and is based on the drawings in Section 1.5 of Rev. 4 of the SAR. The MCNP model includes an explicit description of the fuel tubes in the plenum and top end fitting regions, but does not include modeling of the trunnions. A comparison between the applicant's results and the confirmatory calculations for the cask surface dose rates showed a variation in the results, which is expected when two different codes are used for shielding calculations. For the dose rates at 1 meter from the cask, the applicant's results and the confirmatory calculations are in good agreement. The differences between the applicant's results and confirmatory results fell within acceptable bounds. TS 5.2.3 is based on the applicant's calculations and has been included to specify a maximum allowable dose rate on the surface of the cask at specified locations.

5.4.1.2 Accident Conditions

Table 5.1-2 of the SAR also contains results of calculations for accident-condition dose rates of the design-basis fuel on the cask side, 1 meter from the cask side, the top of the cask, and 1 meter from the top of the cask. Maximum dose rates on the surface and at 1 meter are approximately 1467 mrem/hr and 555 mrem/hr, respectively. Confirmatory calculations for the accident condition were not made explicitly; however, the calculated photon dose rate inside the neutron shield can be used to approximate the applicant's estimate for the accident condition. The staff's confirmatory calculations give a photon dose rate of 860 mrem/hr at the inside boundary of the radial neutron shield. This value is expected to be higher than if the dose rate for the cask were calculated without the radial neutron shield because of radiation return from the radial neutron shield. Considering the affect of the radiation return from the neutron shield, the applicant's photon dose rate of 749 mrem/hr is considered to be within acceptable agreement. Staff found there to be reasonable assurance to accept the accident condition dose rates given in SAR Table 5.1-2.

5.4.1.3 Occupational Exposures

Design-basis fuel at 40,000 MWD/MTU burnup and 10-year cooling time was used to estimate occupational exposures during cask operations. Section 10 of the SAR presents estimated occupational exposures using the calculated dose rates for the locations shown in Figure 5.1-2.

5.4.1.4 Off-Site Dose Calculations

Direct-path off-site dose rates are presented in Table 5.1-3 of the SAR for a single cask. Direct-path dose rates for off-site locations assumed a design-basis fuel loading, level topography, and a 100% occupation time. Confirmatory calculations of the direct dose rates were obtained from the same calculations used to estimate the surface and 1-m dose rates. The applicant's dose rates at 100, 200, 300, and 500 meters are very close to the confirmatory dose rates. Confirmatory skyshine dose rates when a berm or shield is present also were calculated with the MCNP code and are discussed in SER Section 10.

Section 10 of the SER evaluates the overall off-site dose rates from the TN-68 cask. The staff found reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved. The general licensee using the TN-68 cask must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate operational compliance with 10 CFR 72.104(a). The actual doses to individuals beyond the controlled area boundary depend on site-specific conditions such as cask-array configuration, topography, demographics, actual burnup and cooling times of the fuel, and use of engineered shielding features (e.g., berm). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities in the region of the ISFSI such as reactor operations. Consequently, final determination of compliance with 72.104(a) is the responsibility of each general licensee.

The general licensee must also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public, as required by 10 CFR Part 20, Subpart D, by evaluations and measurements.

5.5 Evaluation Findings

F5.1 The SAR sufficiently describes shielding design features and design criteria for the SSCs important to safety.

F5.2 Radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F5.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The TN-68 cask shielding features are designed to assist in meeting these requirements.

F5.4 The staff concludes that the design of the shielding system for the TN-68 is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system provides reasonable assurance that the TN-68 cask will provide safe storage of spent fuel as required by the regulations. This finding is based on a review that considered the specifications in the SAR, the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

5.6 References

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.

2. U.S. Code of Federal Regulations, "Standards for Protection Against Radiation," Title 10, Part 20.

3. Petrie, L.M., et al., "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200, Vol. 1-4, Revision 4, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1995.

4. Croff, A.G., et al., "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code," ORNL/TM-6051, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1978.

5. Luksic, A.T., et al., "The Role of Trace Impurities in Classification of In-Core Reactor Components," EPRI TR-102800, Battelle Pacific Northwest Laboratories, Richland, Washington, 1993.

6. Croff, A.G., "ORIGEN2: A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code," ORNL/TM-5621, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1980.

7. TRW Environmental Safety Systems, Inc., "DOE Characteristics Data Base, User Manual for the CDB_R," November 16, 1992.

8. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200, Vol. 1-3, Revision 5, Oak Ridge, Tennessee, 1997.

9. Briesmeister, J.F., ed., "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4B," LA-12625-M, Los Alamos National Laboratory, Los Alamos, New Mexico, 1997.

6.0 CRITICALITY EVALUATION

The staff reviewed the TN-68 cask criticality analysis to ensure that all credible normal, off-normal and accident conditions have been identified and their potential consequences on criticality considered such that storage of spent fuel in the TN-68 cask meets the following regulatory requirements; 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c) and 72.236(g)[1]. Revision 5 of the SAR was also reviewed to determine whether the TN-68 fulfills the following acceptance criteria listed in Section 6 of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems[2];

a. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95% confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

b. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.

c. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.

d. Criticality safety of the cask system should not rely on the use of the following credits;

- burnup of the fuel,
- fuel-related burnable neutron absorbers,
- more than 75% for fixed neutron absorbers when subject to standard acceptance tests. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed.

6.1 Criticality Design Criteria and Features

The design criterion for criticality safety is that the effective neutron multiplication factor, k_{eff}, including statistical biases and uncertainties, shall not exceed 0.95 for all postulated arrangements of fuel within the cask under normal, off-normal and accident conditions.

The TN-68 cask design features relied upon to prevent criticality are the basket geometry and the fixed neutron poisons in the basket. For the basket, Technical Specification 4.1.1 requires a minimum basket fuel cell opening of 5.97 inches by 5.97 inches and a minimum Boron-10 (^{10}B) areal density of either 0.030 g/cm^2 for the borated aluminum alloy or 0.036 g/cm^2 for the B_4C-aluminum composite material. The applicant took credit for 90% of the minimum specified ^{10}B areal density in the borated aluminum alloy and 75% credit in the B_4C-aluminum composite material. The fabrication requirements and acceptance criteria for the borated aluminum alloy, which justify the use of 90% ^{10}B credit, are outlined in SAR Section 9.1.7. The acceptance criteria require more test samples be taken for the borated aluminum alloy compared to the B4C-aluminum composite material. In addition, neutron transmission and neutron radioscopy or radiography are required on the borated aluminum coupons. These additional testing requirements justify the use of 90% ^{10}B credit for the borated aluminum.

The staff reviewed the TN-68 cask design criteria and features discussed in Sections 1.2, 2.5, and 6 of Revision 5 of the SAR and verified that the design features important to criticality safety are clearly identified and adequately described. The staff verified that the SAR contains engineering drawings, figures, and tables that are sufficiently detailed to support an in-depth staff evaluation.

The staff also verified that the design-basis off-normal and postulated accident events would not have an adverse effect on the design features important to criticality safety. Therefore, based on the information provided in the SAR, the staff concludes that the TN-68 cask design meets the double contingency requirements of 10 CFR 72.124(a).

6.2 Fuel Specification

The TN-68 dry storage cask is designed to store a maximum of 68 General Electric BWR assemblies. The following assembly types (D, C, or S lattice only) are allowed in the cask:

Assembly Size	Designation
7x7	2, 2A, 2B, 3, 3A, 3B
8x8	4, 4A, 4B, 5, 6, 6B, 7, 7B, 8 (with 2 or 4 water rods), 8B (with 2 or 4 water rods), 9, 9B, 10
9x9	11, 13
10x10	12

The fuel assemblies are described in Sections 2.1 and 6.2 of the SAR and the fuel characteristic limits important to criticality safety are given in TS 2.1. The GE designations, along with the lattice specification, describes the mechanical characteristics of the assembly. The maximum uranium mass listed in TS 2.1 conservatively bounds the allowed fuel assemblies. The maximum assembly average burnup is 40,000 MWD/MTU and the maximum lattice-averaged initial fuel enrichment is 3.7 wt% U-235. The lattice-averaged initial enrichment is the average enrichment of the pins across the assembly at any axial plane as shown in SAR Section 6.2. Use of an average axial enrichment or the average bundle enrichment is not allowed. The applicant performed calculations that verify criticality safety is maintained for each of these fuel types in the TN-68 cask. Each fuel type was modeled without a fuel channel, and with fuel channels ranging in thickness from 0.065-inch to 0.120-inch.

Specifications on the fuel condition are also included in Section 6.2 of the SAR and TS 2.1. Fuel with structural defects greater than pinhole leaks and hairline cracks may not be loaded into the TN-68 cask. Fuel bundles with missing pins are not allowed unless the missing pin is replaced by a fuel pin or dummy pin that displaces an equivalent volume.

In Appendix 6A of the SAR, the applicant has shown that a fuel assembly burnup of 40,000 MWD/MTU will not result in fuel cladding failure during the cask drop accidents, which bound all storage conditions. In Section 3 of this SER, the staff reviewed these analyses and agree that the criticality models need only consider intact fuel pins.

The staff reviewed the fuel specifications considered in the criticality analysis and verified that they bound the specifications given in Sections 1 and 2 of the SAR, and the TS. The staff verified that all fuel assembly parameters important to criticality safety have been included in the TS.

6.3 Model Specification

6.3.1 Configuration

A single TN-68 cask infinite in length with full water reflection was modeled in all cases except as discussed below. Although confinement is maintained during all storage conditions and thus prevents any water inleakage, cask loading and unloading operations are typically performed with the cask fully flooded. The active fuel region was modeled explicitly except it was infinite in length. The applicant did not take credit for the burnup of the fuel or for burnable absorbers in the fuel. The basket's borated aluminum poison material which was considered in all scenarios, was generally modeled as the borated aluminum alloy, except as described below. The aluminum spacers between the fuel basket and the cask wall were conservatively modeled with more aluminum and less water than is actually present to reduce the moderation of the neutrons in that region. The applicant showed that the other minor simplifications made in the cask model did not statistically change k_{eff}.

A number of parametric cases were analyzed to determine the most reactive model for normal conditions. First, the results of a uniform enrichment model with all pins containing 3.7 wt% U-235 and variable enrichment models with an average enrichment of 3.7 wt% U-235 were compared for the 7x7, 8x8 and 9x9 assemblies. A total of twenty-three variable enrichment lattices were modeled. The uniform enrichment model results were .0032 ± .0037 higher in k_{eff}. To account for this, the applicant reduced the upper subcritical limit by 4.2 millik (mk) (which is equal to .0032 – (2 x .0037)). No positive biases were applied. The cask model used for these calculations varied slightly from the model used for the criticality analysis. However, the differences were not significant and the results from different models were not compared.

Second, the reactivities of 68 7x7, 8x8, 9x9 and 10x10 assemblies in the cask were compared. A number of fuel generations had the same parameters important to criticality such that one assembly model often bounded more than one fuel generation. Overall, fifteen different assembly models were created. The reactivity of each assembly model was calculated with and without fresh (unborated) water in the fuel-pellet annulus of the fuel rods. The scenarios with water in the fuel rods were always more reactive and all subsequent calculations included this. In addition, the cases were modeled without a fuel channel and with a fuel channel thickness of 0.065-inch, 0.080-inch, 0.0100 and 0.120-inch. The 10x10 GE12 model; the 8x8 GE9, GE9b and GE10 models; and the 7x7 GE2 and GE2b models had the highest reactivities.

The following parametric calculations were performed for the three fuel types with the highest reactivity. The assemblies were shifted toward the center of the cask, which was found to be more reactive than the case with the assemblies centered in the basket compartments. The use of the minimum tolerances on the basket fuel compartment size was more reactive than the use of nominal dimensions. Both of these scenarios increase the interaction between the assemblies in the basket, which is expected to increase the reactivity. The density of the fresh water in the cask was also varied to bound any possible density changes during loading and unloading operations. The full density water resulted in the highest reactivity in all cases. The fuel channel thickness was modeled as 0-inch, 0.065-inch, 0.100-inch and 0.120-inch with the thicker channel being most reactive in all cases. Finally, one case was also modeled with the metal matrix composite absorber which was slightly less reactive that the equivalent model with the borated aluminum absorber.

The normal condition model combined the most reactive conditions from the parametric studies. Thus, the normal condition models for the three most reactive assemblies modeled the fuel assemblies off-center in the basket compartment with minimum basket compartment sizes, full density water and the borated aluminum absorber material. The pellet-clad annulus of the fuel pins contained full density water in all cases.

The accident condition model substituted a single 5 wt% enriched fuel assembly for one of the central 3.7 wt% assemblies in the normal condition model to represent a misloading accident for each of the three most reactive assembly types. The k_{eff} of all accident models was less than the USL of 0.9331 by a minimum margin of 4 mk.

The scenario where the cask is partially filled with unborated water and partially filled with steam was not analyzed, as this will not increase reactivity in this cask design. The interior of the TN-68 cask does not allow for preferential or uneven flooding of the cask, therefore this scenario was also not analyzed.

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in SAR Sections 1 and 2, including engineering drawings. The staff also reviewed the applicant's methods, calculations, and results for determining the worst-case manufacturing tolerance. Based on the information presented in the SAR, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances was incorporated into the calculation models.

The staff performed confirmatory analyses using the information provided in the SAR and TS. Specifically, the staff used Drawing Nos. 972-70-4, Revision 5 and 972-70-5, Revision 4. The staff's fuel assembly models were based on the fuel assembly parameters given in Section 6 of the SAR and TS 2.1. The uranium masses and enrichment were taken from Revision 5 of the SAR as these are the values used in the TS. The staff's results were comparable with those of the applicant.

6.3.2 Material Properties

The compositions and densities for the materials used in the criticality safety analysis computer models are provided in Section 6 of the SAR. The minimum required areal density of the ^{10}B in the fixed neutron poison plates is 30 mg/cm^2 for the borated aluminum alloy absorber and 36 mg/cm^2 for the metal matrix composite absorber. The calculations modeled 90% of the ^{10}B, or 26.9 mg/cm^2 for the borated aluminum absorber and 75% of the ^{10}B, or approximately 27 mg/cm^2 for the metal matrix composite absorber. In SAR Section 9.1.7A, the justification for the use of 90% credit for the borated aluminum material is given, along with acceptance tests for the fabrication of the neutron absorber sheet materials. SAR Section 9.1.7B lists the less rigorous acceptance tests for the metal matrix composite material. See SER Section 9.1.5 for discussions of the qualification and acceptance tests of these two neutron absorber materials for the TN-68 cask design.

The continued efficacy of the neutron absorber plates over a 20-year storage period is assured by the design of the TN-68 cask. Justification for this is given in SAR Sections 6.3.2 and 9.1.7. The neutron absorber is a borated aluminum alloy or metal matrix composite material that is sandwiched between stainless steel tubes and plates that provide the structural support. The fabricated plates meet all thermal requirements and can be expected to have no significant erosion or corrosion under ISFSI service. A structural analysis was performed which demonstrates that the basket plates will remain in place during all accident conditions. The neutron flux in the dry cask over the storage period is also very low such that 10B depletion during 20 years of ISFSI service is negligible. Thus, the staff agrees with the SAR conclusion that the neutron poison will remain effective for the 20-year storage period.

The compositions and densities for the materials in the computer models were reviewed by the staff and determined to be acceptable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant utilized the CSAS modules of the SCALE version 4.3[3] computer codes and the accompanying 27-group cross-section library for the TN-68 cask analysis and the benchmark calculations.

The staff performed confirmatory analysis with the Monte Carlo N-Particle (MCNP) code version 4B developed at Los Alamos National Laboratory[4].

The SCALE and MCNP codes are both standards in the industry for performing criticality analyses. Thus, the staff agrees that the codes and cross-section sets used are appropriate for this particular application and fuel system.

6.4.2 Multiplication Factor

Results of the applicant's criticality analysis show that k_{eff} of the TN-68 cask will remain below 0.95 for all allowed fuel loadings. The staff reviewed the applicant's calculated k_{eff} values and Upper Subcritical Limit (USL) and agrees that these values have been appropriately calculated to include all biases and uncertainties at a 95% confidence level or better.

The staff performed independent calculations using MCNP to confirm the applicant's analysis. Calculations with and without the simplifications in the cask model made by the applicant were performed for the 10x10 fuel assembly. The results showed the simplifications resulted in a higher k_{eff}; thus the applicant's model is bounding. The staff's analysis also confirmed that full density water, the use of minimum compartment sizes and basket tolerances, locating the fuel off-center in the basket compartments towards the center of the cask, and water in the pellet-clad annulus of the fuel, all increase k_{eff}. The only variation between the MCNP confirmatory analysis and the applicant's analysis concerned the most reactive assembly. The MCNP results indicated that the 7x7 GE2 and GE2b assembly was slightly more reactive than the 10x10 assembly whereas the applicant's results indicated the 10x10 assembly had the highest reactivity. However, this type of variation can be expected between two different codes. The applicant performed the accident analysis for both the 7x7 and the 10x10 assemblies to ensure subcriticality of both fuel types in the TN-68 cask. In addition, the applicant adequately benchmarked the SCALE codes against critical experiments and appropriately applied code biases and uncertainties to the SCALE results. Overall, the confirmatory analysis performed by the staff is in close agreement with the applicant's results for the TN-68 cask.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the TN-68 will remain subcritical, with an adequate safety margin, under all credible normal, off-normal, and accident conditions.

6.4.3 Benchmark Comparisons

The applicant performed benchmark comparisons on selected critical experiments that were chosen to bound the variables in the TN-68 cask design. The parameters in the benchmarks bounded the parameters in the analysis with respect to fuel enrichment, fuel pin pitch, boron areal density in the separator plates, hydrogen to U-235 atom ratio, water to fuel volume ratio, assembly separation, and average lethargy causing fission. No significant trends in the bias were found.

The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data and on the same computer hardware used in the criticality calculations.

The staff reviewed the benchmark comparisons in the SAR and agrees that the CSAS module of the SCALE computer codes used for the analysis was adequately benchmarked to representative critical experiments.

An USL of 0.9331 was calculated by the applicant. The USL incorporates the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any k_{eff} less than the USL is less than 0.95, which is the design criterion.

The staff reviewed the applicant's method for determining the USL and found it to be acceptable and conservative. The staff also verified that only biases that increase k_{eff} have been applied.

6.5 Supplemental Information

The following fuel types (D, C, or S lattice only) can be loaded into the TN-68 cask without compromising criticality safety requirements:

Fuel Type (BWR)	Fuel Assembly Designation	Maximum U/assembly	Number of Fuel Rods
GE 7x7	2, 2A, 2B	0.1977	49
GE 7x7	3, 3A, 3B	0.1896	49
GE 8x8	4, 4A, 4B	0.1880	63
GE 8x8	5, 6, 6B, 7, 7B	0.1876	62
GE 8x8	8, 8B	0.1885	62
GE 8x8	8, 8B, 9, 9B, 10	0.1824	60
GE 9x9	11, 13	0.1757	74
GE 10x10	12	0.1857	92

The fuel may have channels up to 0.120-inch thick. The maximum pitch and minimum rod outer diameter are specified in the TS. The maximum lattice-averaged initial fuel enrichment (prior to irradiation) of the assemblies is 3.7 wt% U-235. The lattice-averaged initial enrichment is the average enrichment of the pins across the assembly at any axial plane. Fuel pins with cladding defects greater than pinhole or hairline cracks are not allowed in the TN-68 cask. These fuel pins must be removed from the assembly. All missing fuel pins must be replaced with a fuel pin or a dummy rod with the same external dimensions as the fuel before the assembly can be loaded into the cask.

All supportive information has been provided in the SAR, primarily in Sections 1, 2, 6, and the TS.

6.6 Evaluation Findings

Based on the staff's review of Revision 5 of the TN-68 SAR and the staff's own confirmatory analyses, the staff concludes that the TN-68 cask meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds the following:

F6.1 Structures, systems and components important to criticality safety are described in sufficient detail in Sections 1, 2, and 6 of the SAR and on the design drawings to enable an evaluation of their effectiveness.

F6.2 The TN-68 cask is designed to be subcritical under all credible conditions.

F6.3 The criticality design is based on favorable geometry and fixed neutron poisons in the basket. An appraisal of the fixed neutron poisons has shown that they will remain effective for the 20-year storage period, and there is no credible way to lose them.

F6.4 The analysis and evaluation of the criticality design and performance have demonstrated that the cask will provide for the safe storage of spent fuel for a minimum of 20 years with an adequate margin of safety.

F6.5 The staff concludes that the criticality design features for the TN-68 cask are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the TN-68 cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

6.7 References

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.

2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

3. Scale 4.3, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Oak Ridge National Laboratory, March, 1997.

4. Judith F. Briesmeister, Editor, ``MCNP™ -- A General Monte Carlo N-Particle Transport Code," Los Alamos National Laboratory report LA-12625-M (March, 1997).

7.0 CONFINEMENT EVALUATION

Confinement systems must be designed to ensure that the annual dose equivalent, from normal operations and anticipated occurrences, to individuals beyond the controlled area is less than the limits set forth in 10 CFR 72.104(a)[1]. For design-basis accidents, radiation doses to individuals at or beyond the controlled area must be less than the limits given in 10 CFR 72.106(b). The cask design must also protect the spent fuel cladding against degradation that might lead to gross ruptures as required in 10 CFR 72.122(h)(1). The conclusions in this SER section are based on information provided in TN-68 SAR Revision 5.

7.1 Confinement Design Characteristics

The TN-68 SAR contains a description of the confinement boundary in Sections 1.2.1, 2.3.2, and 7.1 and Figure 1.2-1. The confinement boundary includes the inner shell, the shell bottom plate, shell flange, lid outer plate, lid bolts, vent and drain port cover plates and bolts, and the inner metallic seals on the lid and the vent and drain ports. Confinement welds include a circumferential bottom closure weld, a circumferential weld attaching the top flange to the vessel shell, and longitudinal and circumferential welds needed to construct the cylindrical vessel. The cask lid is bolted to the shell flange on the cask body with 48 bolts. The lid has penetrations for the vent and drain ports that are closed by cover plates attached to the lid by 8 bolts each. The confinement vessel is designed, fabricated, and tested as closely as possible in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB[2]. Exceptions to the ASME Code are discussed in Section 7.1.1 of the SAR and listed in Table 4.1-1 of the TN-68 TS. The staff concludes that the description of the confinement boundary satisfies the requirements of 10 CFR 72.24(c)(3).

Cask closures, including the lid, the vent port cover, and the drain port cover, are sealed by double metallic seals. The double metallic seals, seal materials, the sealing configuration, and seal reliability data are described in Sections 2.3.2.1 and 7.1.3 of the SAR. Each seal is designed to limit leakage rates to much less than the allowable leakage rate (both seals combined) of 1×10^{-5} ref cm^3/sec. Based on the design and test data provided by the applicant, the staff has reasonable assurance that the double metallic seals will provide a reliable and effective seal for spent fuel storage. The staff evaluated information in SAR Section 7.1.3 to demonstrate that the lid seals perform as separate seals when the lid bolts are properly torqued. The staff concludes that the seal design satisfies the requirements of 10 CFR 72.236(e) for redundant sealing of the confinement boundary.

A cask seal overpressure monitoring system (OMS) maintains helium between the inner and outer lid seals and the vent and drain port cover seals (the interseal) at a pressure higher than the cask cavity pressure and atmospheric pressure. As long as the interseal pressure is higher than the cask cavity pressure, cavity gas cannot leak out of the cask and air cannot leak into the cask cavity. The OMS provides continuous monitoring as discussed in SER Section 7.2.

The applicant provided procedure descriptions for draining and vacuum drying the cask interior during loading operations. TS 3.1.1 requires that a pressure of less than 4 millibar (3 Torr) be sustained for at least 30 minutes with the vacuum pump isolated from the system. Removal of water and potentially oxidizing material from the cask is necessary to protect the fuel cladding from degradation during storage. The cask is then backfilled with helium as required by TS 3.1.2. The helium cover gas protects the cladding from oxidation during storage and was also credited in the criticality and thermal analyses. After helium backfill, the cask is sealed and leak tested as required by TS 3.1.3 and 3.1.4 to demonstrate that the total cask leakage rate does not exceed 1×10^{-5} ref cm^3/sec. The staff concludes that these procedure descriptions provide reasonable assurance that residual water and other potentially oxidizing material are removed from the cask and that the fuel will be protected from severe degradation during storage.

The TN-68 uses multiple barriers provided by the fuel cladding and the cask confinement system to assure that there is no release of radioactive material to the environment. Section 3 of the SER concludes that all confinement boundary components, including the lid bolts, are maintained within their code-allowable stress limits during normal, off-normal, and hypothetical accident conditions. SER Section 4 concludes that the peak confinement boundary component temperatures and pressures are within the design-basis limits for normal, off-normal, and hypothetical accident conditions. Leakage rate testing of the lid and vent and drain port cover plate metallic seals assure the integrity of the TN-68 closure. TN described the inspection and test acceptance criteria in SAR Section 9.1. The construction of the TN-68 with the redundant metallic seals, the OMS, and extensive inspection and testing, provides reasonable assurance that release of radioactive material will not occur during normal storage and transfer conditions and that an inert atmosphere will be maintained in the cask cavity over the storage period.

7.2 Confinement Monitoring

The OMS is placed into service at the time of initial fuel storage and maintained over the storage lifetime. The OMS pressurizes the cask interseal region with helium to a pressure above the cask internal pressure. This ensures that helium would leak into the cask cavity if an unanticipated failure of the inner seal occurs. This system of metallic seals and helium pressurization also ensures that air does not leak into the cask cavity, protecting the fuel from degradation.

Pressure transducers or switches in the OMS are designed to be connected to a monitoring panel, provided by the licensee, to signal a low-pressure condition (e.g., OMS pressure below 3.0 atmospheres, absolute (atm abs)). Since the cask cavity is initially backfilled with helium to 2.0 atm abs (+0/-10%) at equilibrium temperature, any leakage would be from the OMS to either the cask cavity or the atmosphere. The minimum 3.0 atm abs limit allows sufficient time to detect and correct problems with cask seals before any potential leakage from the cask cavity occurs. TS 3.1.5 requires monitoring the cask interseal pressure at least once each 7 days and provides for periodic testing of OMS instrumentation.

In SAR Section 7.1.5, the applicant presented an analysis of the OMS pressure verses time, assuming leakage at the tested leakage rate (1×10^{-5} ref cm^3/sec) and temperature change due to decay heat decrease. This analysis demonstrated, and the staff confirmed, that the OMS pressure would remain above the cask cavity pressure and atmospheric pressure. However, if the seals leaked at the assumed tested leak rate, the analysis indicates that the OMS will need to be repressurized after about 10 to 12 years to avoid reaching the minimum 3.0 atm abs limit. Periodic testing of the OMS components per TS Surveillance Requirement 3.1.5.2 provides for verification of proper functioning of the OMS at 3-year intervals. This provides additional assurance that the interseal pressure will be maintained above the cask cavity pressure and thus preclude leakage under normal conditions.

Given the high reliability of the double metallic seals, the simple, reliable design of the OMS, and the required surveillance of the cask interseal pressure, the staff concludes that the OMS meets the requirements of 10 CFR 72.122(h)(4) for continuous monitoring.

7.3 Nuclides with Potential for Release

The quantities of the radionuclides postulated to be released to the environment under off-normal and accident conditions were assessed using the methods and data given in NUREG/CR-6487[3] and ANSI N14.5-1997[4]. The fuel source term used in the calculations consists of all radionuclides that comprise greater than or equal to 0.1% of the total radioactivity in the fuel pins plus iodine, in accordance with NRC Interim Staff Guidance (ISG)-5[5]. In addition, the applicant included selected actinides that were less than 0.1% of the total radioactivity. The source term for Co-60 in the fuel crud was calculated using the method for calculating the inventory of crud on spent fuel surfaces given in NUREG/CR-6487.

The failed fuel and release fractions used in the analysis were taken from NUREG/CR-6487. For off-normal conditions, it was assumed that 10% of the fuel cladding fails and for accident conditions, 100% of the fuel cladding was assumed to fail. The release fractions given in Tables 7.3-1 and 7.3-2 of the SAR were consistent with the release fractions given in NUREG/CR-6487.

The applicant applied an additional 0.10 fraction for release of fuel fines based on the fraction expected to remain airborne after ejection from the fuel rod. This fraction was based on recommendations from Sandia Report SAND90-24066[6]. The staff reviewed this report and experimental results from rod burst tests reported in NUREG/CR-0722[7]. The staff concludes that there is reasonable assurance that the 0.10 fraction bounds the fraction of fuel fines expected to remain airborne in the cask and, therefore, available for release after ejection from the fuel rod.

Leakage rates calculated in Section 7.3 of the SAR were calculated using the methods in ANSI N14.5-1997. The analysis was based on a confinement boundary tested leakage rate of 1×10^{-5} ref cm^3/sec as adjusted for applicable off-normal and accident temperatures and pressures and helium gas properties. The staff performed independent calculations that confirmed the off-normal and accident condition release rates given in the SAR.

7.4 Confinement Analysis

For normal conditions, the staff concludes that no discernable leakage during normal operations is credible and, therefore, dose at the controlled area boundary from atmospheric releases is not calculated because:

- the TN-68 confinement boundary is sealed and leak tested at the time of cask loading,
- the temperature and pressure of the cask are within design-basis limits, and
- the OMS functions to prevent leakage from the cask cavity to the environment.

The applicant evaluated the doses from off-normal and hypothetical accident conditions in the SAR to demonstrate compliance with the applicable requirements for off-normal operations and design-basis accidents. In these evaluations, the OMS pressurization function is assumed to have failed and a leak rate from the cask was calculated for the respective off-normal and hypothetical accident temperatures and pressures using the methodology discussed in SER Section 7.3. Other inputs and assumptions are summarized in SER Table 7-1.

Table 7-1 Summary of TN-68 Confinement Dose Calculation Inputs

Case	% Rods failed	Leak/ exposure duration	Pasquill stability class	wind speed (m/sec)	χ/Q method	Distances to site boundary, meters
Off-normal	10	1 year	D	5	RG 1.145[8]	100, 500
Accident	100	30 days	F	1	RG 1.25[9]	100, 500

TN used dose conversion factors (DCF) from EPA Federal Guidance Reports 11[10] and 12[11]. The staff noted that the applicant used the bounding DCF except for strontium 90 (Sr-90). For Sr-90, the applicant provided an acceptable justification for the use of a lower DCF based on the expected chemical solubility of compounds containing Sr-90. Estimated doses for off-normal and accident conditions assuming a 100 meter site boundary are summarized in SER Table 7-2.

Table 7-2 Summary of TN-68 Confinement Dose Estimates (mrem)

Off-normal Case 100 m	TEDE (limit 25 mrem)	Thyroid (limit 75 mrem)	Lung (limit 25 mrem)	Bone Surface (limit 25 mrem)	Other Organs* (limit 25 mrem)	
	3.43	0.624	13.1	12.1	1.69	
Accident Case 100 m	TEDE (limit 5000 mrem)	Thyroid (limit 50000 mrem)	Lung (limit 50000 mrem)	Bone Surface (limit 50000 mrem)	Other Organs* (limit 50000 mrem)	Skin (limit 50000 mrem)
	92.4	14.1	307	404	42.2	0.89

* Includes remaining organ tissue except skin and lens of the eye (e.g., liver, spleen, brain, and large and small intestines).

By review of the applicant's calculations and independent confirmatory calculations, the staff confirmed that the applicant's results and methods for estimating doses from postulated releases were consistent with the SRP, NUREG-1536[12], and ISG-5. Compliance with the dose-equivalent limit for the lens is achieved here by demonstrating compliance with the dose-equivalent limit for the skin and the effective dose-equivalent limit, consistent with guidance in ICRP-26[13]. The estimated doses for the minimum site boundary of 100 meters and beyond are within the limits of 10 CFR 72.104 for off-normal conditions and 10 CFR 72.106 for accident conditions, and are acceptable.

7.5 Latent Seal Failure Evaluation

As previously discussed, the TN-68 seals are designed for high reliability and analyzed to maintain integrity during normal, off-normal, and design-basis accident conditions during the licensed lifetime. The OMS allows for detection of postulated gross seal leakage within the frequency of the surveillance requirements. However, for postulated seal leakage rates greater than the tested rate, but not gross leakage, there could be a lag time before OMS pressure decays to 3.0 atm abs and indicates a low pressure condition. This degraded seal leakage is considered a "latent" condition and should be presumed to exist concurrently with other off-normal and design-basis events.

For the off-normal case, the OMS will limit leakage to the tested rate and, therefore, the results of the off-normal analysis address the latent condition. The OMS system is not designed to withstand design-basis accident conditions and, therefore, its function is not considered for the hypothetical accident concurrent with a latent degraded seal condition.

The applicant provided the results of sensitivity studies in SAR Section 7.3.3 to determine (1) the delay time from the onset of the latent condition to the point where the OMS would indicate system leakage and (2) the dose consequences if an accident occurred concurrent with the latent condition. For a leak rate of $1x10^{-3}$ ref cm^3/sec (100 times the tested leak rate) the delay time to indication of a degraded seal was 16 days. At this leak rate, the applicant's dose assessment concluded that the dose limits of 10 CFR 72.106(b) could be exceeded after the leak condition existed for 16 days.

The staff has reasonable assurance that the design of the TN-68 for latent seal leakage conditions is acceptable based on the following considerations:

- The possibility of the occurrence of a design-basis event that would remove the OMS concurrent with a degraded seal condition is judged to be very remote. The short delay time for detection of the latent condition further reduces the possibility of occurrence.

- If the accident were to occur, the staff expects that actions to mitigate the event would occur in less than 16 days. In SAR Section 8.4, the applicant has described recovery actions including installation of a blind flange on the OMS port to mitigate this event.

- In its dose assessment, TN calculated atmospheric dispersion factors (χ/Q) using the guidance of RG 1.25[14]. While the χ/Q model used in RG 1.25 provides a bounding estimate of the χ/Q, the model is applicable to short-term release durations (the assumed duration was 2 hours). An updated model for estimating χ/Q for longer release periods is provided in RG 1.145[15]. The staff performed independent dose calculations using χ/Q values based on the RG 1.145 model and found that the dose limits would not be exceeded for a 30 day accident with leak rates of 100 times the tested leak rate.

7.6 Evaluation Findings

F7.1 Sections 1, 2, and 7 of the SAR describe confinement SSCs important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the TN-68 adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the SER discusses the staff's relevant temperature considerations.

F7.3 The design of the TN-68 provides redundant sealing of the confinement system closure joints using double metallic seals and a bolted lid. Penetrations into the cask cavity include a vent and drain port, both of which are in the cask lid. Both penetrations are sealed with double metallic seals and bolted closures.

F7.4 The confinement system is monitored with an OMS system as described in Section 7.2 of the SER . No instrumentation is required to remain operational under accident conditions.

F7.5 The quantity of radioactive materials postulated to be released to the environment has been assessed as discussed above. In Section 10 of the SER, the dose from these releases is added to the dose from direct radiation to demonstrate that the TN-68 design satisfies the requirements of 10 CFR 72.104(a) and 72.106(b).

F7.6 The cask confinement system has been evaluated by analysis to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and hypothetical accident conditions.

F7.7 The staff concludes that the design of the confinement system of the TN-68 is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the TN-68 will allow safe storage of spent fuel. This finding is reached based on reviews that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant's analysis and the staff's confirmatory analysis, and accepted engineering practices.

7.7 References

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste," Title 10, Part 72.

2. ASME Boiler and Pressure Vessel Code, Section III, Division I, 1995 including 1996 addenda.

3. Anderson, B.L., R.W. Carlson, and L.E. Fischer, "Containment Analysis for Type B Packages Used to Ship Various Contents," NUREG/CR-6487, Lawrence Livermore National Laboratory, Livermore, California, November 1996.

4. American National Standards Institute (ANSI), "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," ANSI N14.5-1997, New York, New York, February 1998.

5. Brach, E. W., "Spent Fuel Project Office Interim Staff Guidance 5," ISG-5, Revision 1, U. S. Nuclear Regulatory Commission, May 1999.

6. Sanders, T. L., et al., "A Method for Determining the Spent Fuel Contribution to Transport Cask Containment Requirements," SAND90-2406, Sandia National Laboratories, Albuquerque, New Mexico, November 1992.

7. Lorentz, R. A., et al., "Fission Product Release from Highly Irradiated LWR Fuel," NUREG/CR-0722, Oak Ridge National Laboratory, Oak Ridge, Tennessee, May 1980.

8. U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1983.

9. U.S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating Accidents in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors," Regulatory Guide 1.25, March 1972.

10. U.S. Environmental Protection Agency (EPA), "Limiting Values of Radionuclide Intake and Air Concentration Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report No. 11, EPA 520/1-88-020, Washington, DC, September 1988.

11. U.S. Environmental Protection Agency (EPA), "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, EPA 402-R-93-081, Washington, DC, September 1993.

12. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

13. International Commission on Radiation Protection, "Statement from the1980 Meeting of the ICRP," ICRP Publication 26, Pergammon Press, New York, New York, 1980.

8.0 CASK OPERATIONS

The staff reviews descriptions of cask operations to ensure that the applicant's SAR presents acceptable operating sequences, guidance, and generic process controls for three key operations: cask loading and handling, cask storage operations, and cask unloading.

The information provided in Section 8 of the SAR, Revision 5, forms the basis of the staff conclusions in this SER Section.

8.1 Cask Loading and Handling

The TN-68 SAR describes generic cask loading operations. Detailed cask loading procedures must be developed by each cask user. Based on the information in SAR Section 8, as discussed below, the staff concludes that the cask loading descriptions provide an adequate basis for the development of the more detailed site-specific operations and test procedures. In addition, the staff concludes that the TN-68 cask is compatible with wet loading. The staff also concludes that the cask loading descriptions presented in the SAR are in the proper sequence and are of sufficient detail that cask users will be able to develop detailed site-specific procedures that adequately protect the workers, public, and the environment and will protect the fuel from significant damage or degradation.

8.1.1 Cask Preparation

The cask loading descriptions presented in SAR Section 8 include important prerequisite, preparation, and receipt inspection provisions to prepare the cask for loading. Preparations include visual inspections of important components for damage such as the cask sealing components and closure bolts. The loading descriptions include steps to replace the lid, vent port, and drain port seals.

8.1.2 Fuel Specifications

The cask loading descriptions in the SAR state that pre-selected fuel assemblies may be loaded into the cask basket. Table 8.1-1, Section B, of the SAR also states that a procedure shall be developed by the user to ensure that the fuel loaded into the cask meets the fuel specifications and that the identity of the fuel assemblies loaded into the cask shall be verified. The fuel assembly specifications for the TN-68 cask are provided in TS 2.1. The site-specific procedures to be developed by each cask user are subject to evaluation at each site through the inspection process. The staff concludes that the cask loading descriptions and TS requirements provide an acceptable means to ensure that fuel loaded in the TN-68 cask will meet the fuel-related assumptions (e.g., inventory, heat load, criticality-related parameters) made in the TN-68 SAR analyses.

8.1.3 ALARA

The staff concludes that the TN-68 cask loading descriptions adequately incorporate general as low as reasonably achievable (ALARA) principles and practices. The loading descriptions provide for a radiation survey to ensure the external gamma and neutron dose rates are below limits and for decontamination of the external surfaces of the cask until acceptable levels of contamination are obtained. These procedure actions are in conformance with TS 3.2.1 and 5.2.3. The smooth external surfaces of the TN-68 facilitate decontamination. The cask loading descriptions incorporate notes to indicate elevated dose rates, provisions for temporary shielding, and other ALARA practices during loading.

Any radioactive effluents generated during cask loading will be governed by the 10 CFR Part 50 license conditions.

8.1.4 Draining and Drying

Based on the discussion below, the staff concludes that the SAR describes acceptable methods for draining and drying the cask. The main intent of the draining and drying operations is to remove water and oxidizing impurities from the cask cavity to protect the fuel cladding from degradation.

The TN-68 lid is placed on the cask while it is submerged in the pool. Cask loading descriptions require several bolts to be installed and hand tightened above the water surface of the pool, prior to fully removing the cask from the pool. After the cask is removed from the pool, the remaining lid bolts are installed and torqued to the values specified in SAR Drawing 972-70-1. Verification of proper bolt torque is also required. Similarly, bolt torque requirements for the vent and drain port covers are provided on Drawing 972-70-1.

When the lid is placed on the cask, the SAR description provides for cask venting to preclude inadvertent pressurization as water in the cavity heats up. The bulk of the water is pumped from the cask via the drain line, and a vacuum drying system is then used to remove residual water from the cask cavity. Precautions are given to either control the evacuation rate, or provide a heat source on the evacuation line, to prevent blockage of the line by ice. Cask pressure is reduced to 4 millibar (3 Torr) and held for at least 30 minutes to verify appropriate levels of dryness are achieved. If the pressure increases above 4 millibar during the 30-minute holding time, the vacuum pumping process is repeated until this criterion is met. These steps are consistent with TS 3.1.1.

8.1.5 Filling and Pressurization

After vacuum drying, the TN-68 cask is backfilled with helium to slightly above atmospheric pressure and the vacuum drying adapter is replaced by a quick disconnect fitting. The cask is then re-evacuated to a pressure of 100 millibar and refilled with helium to a minimum pressure of 2.0 atm abs (+0/-10%) as specified in TS 3.1.2. A minimum helium purity of 99.99% is specified in Section 8.1.3 of the SAR and TS 4.1.4. Calculations presented by the applicant indicated this process will leave about 0.20 gram-mole of oxidizing impurities in the cask cavity. This is less than the 1 gram-mole/cask recommendation given in PNL-6365[1] and is therefore sufficient to prevent severe fuel degradation during the 20-year storage period. Independent staff calculations of the residual impurity levels agree with the applicant's results. The SAR also states that the evacuation and backfill process must be repeated if the cask cavity is exposed to the atmosphere.

8.1.6 Cask Sealing

Section 8 of the SAR describes steps to properly seal the cask, including helium backfill, necessary bolt torque, and leak testing. The steps for properly placing and tightening the lid, drain port, and vent port cover bolts are consistent with the analyses presented in SAR Sections 2 (design criteria), 3 (structural evaluation), and 9 (acceptance tests and maintenance program).

The cask is leak tested using helium mass spectrometry after being backfilled with helium. Leak test methods are consistent with ANSI/ANS N14.5-1997[2], as stated in SAR Section 9. The combined leak rate for all closure seals and the overpressure system is required by TS 3.1.4 to be less than 1×10^{-5} ref cm^3/sec. The SAR includes steps for final installation and testing of the OMS and corrective actions for a failed leak test that include returning the cask to the pool to replacing the lid seal. The staff concludes that the cask sealing, leak test, and corrective actions described in the SAR provide an acceptable basis for development of site-specific procedures.

8.2 Cask Handling and Storage

8.2.1 Cask Handling

All accidents applicable to the transfer of the cask to the storage location are bounded by the design events identified and evaluated in Sections 2 and 11 of the TN-68 SAR. The structural (Section 3) and thermal (Section 4) evaluations presented in the SAR bound conditions that could potentially be created during cask lifting and transfer operations. TS 3.1.6 limits cask lifting if the outer surface of the cask is below -20°F. For cask transport operations, the cask lift height above the transport surface will generally be limited to less than 18 inches. In addition, TS 5.2.2 requires that a site-specific transport evaluation program be developed to evaluate transport route conditions to ensure that design-basis drop limits are met. Consistent with TS 4.2.1, cask handling operations ensure that the casks are spaced a minimum of 16 ft apart, center-to-center, to ensure adequate spacing as assumed in the thermal analysis. The staff concludes that the SAR descriptions of cask handling provide a sufficient basis for development of detailed site-specific procedures.

8.2.2 Cask Storage

Surveillance and maintenance requirements during the storage period are described in SAR Section 8.3 in sufficient detail to permit cask users to develop detailed procedures. Appropriate descriptions and precautionary statements are provided for maintaining and testing the overpressure system. Maintenance operations, discussed in SAR Section 9, are anticipated to be minimal over the lifetime of the cask. Verification that the interseal pressure exceeds 3.0 atm is performed every 7 days and periodic testing of the overpressure instrumentation is performed in accordance with TS 3.1.5. The staff concludes that descriptions of the inspection, surveillance, and maintenance operations provide an adequate basis for development of detailed procedures by cask users.

There will be no routine radioactive effluents generated during storage operations. Gaseous, liquid, and particulate releases from the cask cavity are not anticipated due to the metallic seals and overpressure system. The external surfaces of the cask are decontaminated before it is transported to its storage location, so no significant contamination of the storage area is anticipated. Routine surveillance and maintenance activities do not introduce the potential for radioactive contamination. As a result, the staff concludes that no significant radioactive effluents are generated during storage operations.

8.3 Cask Unloading

As with the cask loading, each cask user will be required to develop site-specific cask unloading procedures. The basis for the detailed user-developed cask unloading procedures is provided in Section 8.2 and Table 8.2-1 of the SAR. The general actions to be taken during unloading include transferring the cask to the spent fuel building, sampling the cask cavity gas, connecting fill and drain lines, lowering the cask into the pool, reflooding the cask with water, removing the cask lid, and removing the fuel assemblies from the storage basket. Several precautions are described to ensure that personnel are adequately protected during unloading operations. The staff concludes that the TN-68 cask is compatible with wet unloading. In addition, the staff concludes that the description of cask unloading operations presented in the TN-68 SAR will provide a sufficient basis for development of safe and effective detailed site-specific procedures.

8.3.1 Damaged Fuel

The SAR describes appropriate contingency actions to be taken prior to lid removal to detect damaged or degraded fuel in the cask. Degraded fuel would be detected via a cavity gas sample taken from the vent port. If degraded fuel conditions are suspected, additional measures are to be taken to prevent personnel contamination or exposure to airborne radioactive materials. The SAR indicates that the special precautions are to be planned, reviewed, and approved by the cask user's designated approval authority. The requirement for cover gas sampling prior to lid removal, and the special precautions provided are acceptable to the staff.

8.3.2 Cooling, Venting, and Reflooding

If the cover gas sample indicates the fuel is not degraded, the helium in the cask cavity is depressurized to atmospheric pressure, fill and drain lines are attached to the fill and drain ports in the cask lid, and the cask is lowered into the spent fuel pool. A typical vent and fill arrangement is shown in SAR Figure 8.2-1. The unloading procedure cautions cask users to ensure that the fill and drain lines are designed for steam at 100 psig to protect against failures that could result in radiological exposures as well as personnel hazards (e.g., steam burns). Water is slowly added through the drain port to fill the cask and gradually cool the fuel.

An analysis of cask pressure during reflood operations was presented in SAR Section 4 to demonstrate that cask pressures remain below the 100 psig design pressure limit. This analysis is the basis for controlling cask inlet water flow rates to 1 gallon per minute or less during the initial phase of cask fill. As stated in the SAR Section 8.2, operators are to close the water supply line inlet valve if the pressure reaches 70 psia (55.3 psig). A valve will be installed at the inlet to the cask to restrict cooling water flow if cask pressure exceeds inlet water pressure (90 psia maximum). Cask users must develop site-specific reflood procedures that control fill rates to ensure that the design pressure of the cask is not exceeded. The staff concludes that actions to prevent cask overpressurization were acceptable.

8.3.3 Fuel Crud

The SAR descriptions of unloading operations incorporate precautions and steps to prevent or mitigate the potential dispersal of fuel crud particulate material. These include a cover gas sample prior to lid removal and monitoring the water/steam mixture ejected from the vent port discharge during reflood operations. The applicant provided a note in the SAR to alert cask users to the possibility that fuel crud could cause an airborne or direct radiation hazard due to floating particulates on the pool surface. The applicant provided suggested crud contamination control measures, including enhanced fuel pool filtration, increased area ventilation, and increased monitoring. The procedures and cautions regarding fuel crud were acceptable to the staff.

8.3.4 ALARA

The TN-68 cask unloading descriptions incorporate general ALARA principles. ALARA practices include provisions to sample cask cavity gases to identify potential fuel cladding damage, monitoring of the water/steam ejected from the vent line during reflood, temporary radiation shielding, and respiratory protection, where necessary. ALARA principles are also reflected in various warnings and notes included in the procedures. Each cask user will need to develop detailed unloading procedures that reflect the ALARA objectives of their site-specific radiation protection programs. The staff concludes that ALARA principles were adequately addressed in the TN-68 cask unloading procedures.

Any radioactive effluents generated during cask unloading will be governed by the 10 CFR Part 50 license conditions.

8.4 Evaluation Findings

F8.1 The TN-68 is compatible with wet loading and unloading. General descriptions for these operations are summarized in Section 8 of the SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.2 The bolted lids of the cask allow ready retrieval of the spent fuel for further processing or disposal as required.

F8.3 The smooth surface of the cask is designed to facilitate decontamination. Only routine decontamination will be necessary after the cask is removed from the spent fuel pool.

F8.4 No significant radioactive waste is generated during operations associated with the ISFSI. Contaminated water from the spent fuel pool will be governed by the 10 CFR Part 50 license conditions.

F8.5 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during cask loading and unloading will be governed by the 10 CFR Part 50 license conditions.

F8.6 The general cask operations described in the SAR are adequate to protect health and minimize damage to life and property. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.7 Section 10 of this SER assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

F8.8 The staff concludes that the generic guidance for the operation of the TN-68 are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the cask operation descriptions provided in the SAR offers reasonable assurances that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

8.5 References

1. Knoll, R.W., and E.R. Gilbert, "Evaluation of Cover Gas Impurities and their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, Pacific Northwest Laboratory, Richland, Washington, November 1987.

2. American National Standards Institute (ANSI), "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," ANSI N14.5-1997, New York, New York, February 1997.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The staff reviewed the acceptance tests and maintenance program in the TN-68 SAR to ensure they are appropriate and that the applicable acceptance criteria have been satisfied in compliance with 10 CFR Part 72[1]. The principal objective of the acceptance tests and maintenance programs is to support commitments for TN-68 dry storage casks. A clear, specific listing of acceptance test and maintenance program commitments helps avoid ambiguities concerning design, fabrication, and operational testing requirements when the NRC staff conducts subsequent inspections.

9.1 Acceptance Tests

9.1.1 Visual and Nondestructive Examination Inspections

Visual inspections are performed at the fabricator's facility to ensure that the cask materials and components conform to the drawings and specifications, all specified coatings are applied, and the cask is free of defects. The casks are also visually inspected upon arrival at the user's facility to ensure the casks were not damaged during shipment and that casks are in conformance with the drawings and specifications. Any defects detected at the user's facility will be repaired or evaluated with respect to the effects of the defect on the component's ability to perform it's intended safety function. Cask design, fabrication, and testing are performed in accordance with the requirements of 10 CFR 72, Subpart G - Quality Assurance.

The TN-68 confinement boundary welds are designed, fabricated, tested, and inspected in accordance with the ASME Boiler and Pressure Vessel Code Subsection NB[2]. Nondestructive examination (NDE) requirements for welds are specified on the drawings provided in the SAR Section 1 using standard NDE symbols and notations in accordance with American Welding Society (AWS) standard 2.4[3]. Exceptions to the ASME Code are specified in Section 7 of the SAR and Section 4 of the TS. Circumferential and longitudinal confinement boundary welds are examined volumetrically by radiography and liquid penetrant or magnetic particle methods and accepted in accordance with ASME NB-5000 standards. The bottom inner plate weld will be inspected using ultrasonic examination methods if the weld is applied before the outer and inner shells are assembled. This inspection is done radiographically using either liquid penetrant or magnetic particle methods if the weld is applied after assembly. Non-confinement welds are inspected in accordance with the ASME Code, Subsection NF. Additional inspections will also be performed on the gamma shield shell to the bottom shield weld and the lid to the shield lid weld as specified in the SAR, Section 3E. NDE personnel are qualified in accordance with American Society for Nondestructive Testing Recommended Practice SNT-TC-1A[4].

With exceptions noted in Section 3.1.2.3 of the SAR, basket welds are fabricated and inspected in accordance with ASME Code, Subsection NG. Inspections include progressive dye penetrant and 100% visual inspection methods, assisted by remote visual inspection using mirrors and auxiliary lighting for basket welds that are not directly visible. Mechanical testing will be performed on at least one coupon from each welding machine to verify proper machine settings and operation prior to each working shift. Acceptance criteria for each weld test are based on failure of the base metal, prior to failure of the weld area, and visual verification that the fused weld zone is 0.5-inch in diameter. In addition, bubble leak tests are performed at 4.5 psi or greater on the resin enclosure to identify leak passages on the weld enclosures.

All structural materials are chemically and physically tested to ensure that the required properties are met. The confinement vessel materials are impact tested in accordance with the ASME Code Section III, Subsection NB, paragraph NB-2300, and meet the acceptance standards in paragraph NB-2330. Ultrasonic examinations of the closure flange and other forgings that form part of the confinement boundary are performed in accordance with paragraph NB-2542 and the acceptance standards provided in paragraph NB-2542.2. All external and accessible internal surfaces are tested using the liquid penetrant or magnetic particle methods in accordance with paragraph NB-2546 or NB-2545. Acceptance standards presented in paragraphs NB-2546.3 and NB-2545.3 are applied. Lid bolts, vent and drain cover bolts, and holes for the bolts are visually inspected in accordance with NB-2582. The lid bolts are also dye penetrate tested in accordance with NB-2520.

The plate and forging materials of the confinement boundary will be examined by ultrasonic methods in accordance with ASME Code, Section III, Subsection NB, paragraphs NB-2530 and NB-2540, respectively. External and accessible internal surfaces of the forging materials will be examined by liquid penetrate or magnetic particle methods in accordance with NB-2546 or NB-2545. Welds will be examined by radiographic methods and either liquid penetrate or magnetic particle methods in accordance with Subsection NB, paragraphs NB-5210, NB-5220, and NB-5230. Allowable surface and subsurface flaw sizes are given in SAR Appendix 3E. Fracture toughness testing is discussed in SER Section 9.1.2.

The applicant determined allowable flaw sizes for the gamma shield material (see SAR Appendix 3E) and committed to perform dye penetrate or magnetic particle testing on the final welds. No special examination requirements are specified for the gamma shield forged steel and plate components because the allowable flaw size is larger than the flaws generally observed in forged steel and plate components.

The NRC staff concludes that appropriate standards have been cited for the visual and nondestructive examination inspections to be performed on the TN-68, including ASME Code Section III, Subsections NB, NF, and NG, as stated in the SAR Sections 7 and 9. The welds and NDE requirements are clearly stated on the drawings in Section 1 of the SAR using appropriate AWS symbols.

9.1.2 Structural and Pressure Tests

The TN-68 cask design uses components that are subject to brittle fracture at low temperatures of service. In accordance with guidance in NUREG/CF-1815, Section 5.1.1, the NRC established two methods for identifying suitable materials: These permit the use results of T_{NDT} determinations (ASTM E-208), and CVN tests (ASTM E-23).

Fracture toughness of the TN-68 confinement boundary components is ensured by material selection and testing. The required nil-ductility transition temperature of TN-68 cask materials is -80°F, which is 60°F below a service temperature of -20°F. Confinement boundary components will be tested in accordance with ASME Code, Section III, Division I, Article NB-2330. In addition to determining the T_{NDT}, CVN testing will be performed at a temperature no greater than 60°F above the T_{NDT} temperature. More details are given in SER Section 3. Acceptance criteria are that, at this temperature, in CVN tests the material shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lbs absorbed energy. The NRC staff concludes that this material selection and testing provides reasonable assurance that the confinement boundary materials will not be susceptible to brittle fracture at -20°F, which corresponds to the lowest (averaged throughout the day) temperature for which the cask has been approved for service.

The applicant conducted a similar analysis to determine testing requirements and acceptance criteria for the gamma shield material. Preliminary Charpy data provided by the manufacturer indicates the gamma shield material has relatively good Charpy impact properties at -20°F, with an allowable flaw size larger than the flaws generally observed in forged steel and plate components. The NRC staff concludes that this material selection is likewise adequate for the requirements of the gamma shield material.

Subsections of structural and pressure tests are given below for lifting trunnions and for hydrostatic testing. Other tests, i.e., tests related to leaks, shielding, neutron absorbers, and thermal considerations, are discussed in separate sections.

9.1.2.1 Lifting Trunnions

To ensure that the lifting trunnions perform satisfactorily, the trunnions on the TN-68 are load tested at three times the design lift load for 10 minutes in accordance with ANSI N14.6[5]. Following the load tests, the trunnion weld and bearing surfaces are examined using liquid penetrate or magnetic particle examination methods. Acceptance standards for these inspections are in accordance with ASME Code, Section III, Articles NF-5340 and NF-5350. NDE personnel are qualified in accordance with SNT-TC-1A. These tests are acceptable to the NRC staff.

9.1.2.2 Hydrostatic Testing

A hydrostatic test of the confinement vessel, at 1.25 times the maximum operating pressure of 100 psig, will be performed in accordance with ASME Code, Section III, Article NB-6200 or NB-6300, except that the confinement vessel is installed in the gamma shield during testing. All confinement welds are fully radiographed in accordance with ASME Code, Subsection NB requirements. The stresses due to internal pressure are small when compared with the confinement boundary loads caused by the drop and tipover events. The NRC staff agrees that the results of hydrostatic test, structural analysis, and radiographic examinations of all confinement boundary welds provide reasonable assurance that the confinement boundary components can adequately withstand the effects of internal pressure.

9.1.3 Leak Tests

The applicant states that leakage tests are performed on the confinement system and overpressure system at the fabricator's facility using the helium mass spectrometry method or other method that provides the required sensitivity. Leakage tests are performed in accordance with ANSI N14.5[6]. The total leak rate, across the lid, vent, and drain port seals, must be shown to be less than 1×10^{-5} ref cm^3/sec at standard conditions and the sensitivity of the leak test procedure must be at least 5×10^{-6} ref cm^3/sec. Similar leakage rate criteria and sensitivities are applied to the overpressure system. The NRC staff concludes that the applicant's leak test requirements are in accordance with established requirements and are acceptable.

9.1.4 Shielding Tests

The neutron shield consists of a poured resin material, a proprietary borated reinforced polymer. Qualification testing of the procedures and personnel used for mixing and pouring the neutron shield will be performed to ensure that its properties meet the required specifications. During fabrication, both composition (chemical analysis) and density of the resin will be periodically tested to ensure consistency. The process controls include appropriate measures to ensure the absence of large voids. External dose rate surveys are performed on the loaded casks, as a final verification of neutron shield performance. TS 5.2.3 is included to limit the dose rate at the cask surface. The NRC staff concurs that together, these measures and specifications provide reasonable assurance that the cask will provide adequate shielding.

9.1.5 Neutron Absorber Tests

Two different plate materials are approved for use as the neutron absorber in the TN-68 cask design. These are a borated aluminum material and a specific composition of a metal matrix composite material called Boralyn (TM). These neutron absorber materials are used to ensure subcriticality during loading and unloading operations that use deionized water inside the vessel. Structural credit is not assigned to these materials in the cask safety analysis, but the thermal conductivity of these materials is considered in the thermal analysis.

9.1.5.1 Borated Aluminum

The borated aluminum plate material is a wrought aluminum alloy that contains 1.7 wt% boron that has been enriched to 95 wt% ^{10}B prior to being used as an alloy addition to the borated aluminum. In the finished product, this enriched boron is in the form of boride precipitates of AlB_2 and TiB_2. Apart from the boron and these precipitates, the matrix is a standard aluminum alloy limited to any in the 1000 series, or either alloy 6063 or 6351. These alloys may also contain titanium, which when added will decrease the average size of the precipitates. The boride precipitates are very stable, second-phase particles that are very finely dispersed, with diameters in the range of 1 to 10 micrometers (μm). The effects that these second-phase particles are expected to have on the physical properties related to the durability of the parent alloy in service are minimal, as the particles are uniform, finely dispersed, inert, equiaxed, second-phase particles. As these precipitates are stable and durable, the durability of the borated aluminum plate material is expected to be governed by, and to be similar to, that of the matrix of the aluminum alloy. The service conditions (radiation and thermal) are not so severe as to promote significant alterations of the aluminum alloys used for this neutron absorbing material during the 20 year licensing period. Therefore, durability of these neutron absorbing materials is regarded to meet or exceed the service requirements of this application.

It is noted that, in production of the plate material, finished plates are visually examined for significant imperfections which can be removed if doing so does not result in a dimensional non-conformance. Such a non-conformance could render the effective boron content of the plate to be inadequate, i.e. less than the required content.

The significant variable in the borated aluminum plate material, which relates to service performance in this application, is the required areal density (30 mg/cm^2) of ^{10}B, which is largely present as boride precipitates. The ^{10}B content determines the effectiveness of the plate material as a neutron absorber. Samples taken from these plates are used to measure both the ^{10}B content and its uniformity. Coupon samples are taken from the plate material, as described in Section 9.1.7 and shown in Figure 9.1-1 of the SAR. On these coupons, the ^{10}B content is measured by neutron transmission tests. The results of these measurements are compared with appropriate standards (e.g., ZrB_2 or TiB_2) and used to verify that the ^{10}B content meets the value required for the application. The effective boron content of each coupon, minus 3σ based on neutron counting statistics for that coupon, must be greater than or equal to the required minimum value of 30 mg $^{10}B/cm^2$ of plate material. Adequate procedures have been established to handle the case in which a coupon fails this test. In addition to the verification of the ^{10}B content, the uniformity of the distribution of the ^{10}B is verified by neutron radioscopy or radiography of the coupons. Acceptance is based on uniform luminance across the coupons. These methods of testing the effective boron content and its uniformity in the plate material are regarded to be sufficient for verification that the neutron absorbing material is present at the required level and in a distribution that will satisfy the requirements of this application. The applicant takes credit for 90% of the 30 mg $^{10}B/cm^2$ and the staff concurs that this is justified based upon data on the effective ^{10}B content and the uniformity of the boron in the plate material.

9.1.5.2 Metal Matrix Composites and Boralyn (TM)

The other type of neutron absorber material approved for the TN-68 storage cask design is a metal matrix composite plate material produced from powders. On the basis of a review of the qualification test data results, a specific composition of a product called Boralyn(TM) is approved for use in the TN-68 cask design. Boralyn(TM) is the trademark for a compact containing a ceramic reinforcement in an aluminum alloy. The specific Boralyn(TM) composite approved for this application is designated 1100/B4C/15p. This compact material is produced using powder particles of a single phased 1100 aluminum alloy matrix and 15 volume percent B_4C particles of an average size of 15 µm with a range of from 1 to 25 µm. *[A 1000 series alloy of purity greater than that of the 1100 alloy would also be acceptable as it would pose no theoretical reason to be less durable as a matrix for this application.]* Plates of this approved material are produced by hot vacuum pressing of blended powders in a billet that is extruded and cut into preforms that are subsequently cross rolled into a plate that is cut to sizes required by the TN-68 design.

As shown in Figure 9.1-2 of the SAR, acceptance tests for these plates use coupon samples taken from every other set of the plates that are produced from the preforms. Neutron transmission measurements are taken either on the test coupons or along the edge of a plate to measure the effective ^{10}B content. As described in SAR Section 9.1.7, the acceptance tests require a minimum areal density of 36 mg/cm^2 of ^{10}B, which is less than the calculated amount in a composite containing 15 volume percent B_4C. The safety margins in the criticality safety evaluation require a minimum ^{10}B areal density of 27 mg/cm^2 for this material. Due to the potential for inhomogeneity, only 75 percent of the specified minimum areal density of 36 mg/cm^2 of ^{10}B is credited to the calculated effective ^{10}B content, so as to ensure that the required minimum will always be present despite any uncertainties. Other acceptance tests include visual examinations of plate quality and measurements of thermal conductivity.

Qualification tests are conducted at least once, for a given set of materials and manufacturing processes, to demonstrate the acceptability of the resulting product as neutron absorber plates. For Boralyn(TM) numerous tests were conducted to obtain reasonable assurance of performance. These included, as for example, tests to demonstrate that the following characteristics and requirements were met for Boralyn(TM):

- The material shall not contain significant voids, shall have near-theoretical density and shall exhibit a uniform distribution, of B_4C particles in an aluminum alloy matrix with no apparent banding or swirling patterns or regions of abnormal (low or high) concentrations of particles, as viewed at magnifications of 50X to 150X or as determined by equivalent optical or other methodologies.

- Uniformity of the B_4C over an entire plate shall be demonstrated using neutron radiography or radioscopy on at least one absorber plate that was randomly selected from a lot produced identically to that proposed for use in production.
- The material shall not exhibit reaction products that could affect service performance. For example, B_4C boundaries should appear clean and free of fracturing and void spaces. The matrix grain boundaries shall exhibit qualities associated with good sintering practice, exhibiting neither oxide-coated aluminum particles, nor substantive reactions between the B_4C and the aluminum. The structure should exhibit minimal contamination from iron and its compounds/phases.
- Durability shall be demonstrated under thermal and radiation conditions that simulate anticipated service conditions. These tests will establish that service exposure will not lead to any formations of significant voids, fracturing, or adverse reaction products that could either compromise the safety related characteristics as a neutron absorber plate material or seriously deteriorate the mechanical properties.
- Durability tests may treat temperature and fluence as independent variables but when this is done, the separate tests (thermal and fluence) should be conducted over fluence levels and for periods at temperature that exceed, with comfortable margins, the conditions to be encountered over the expected service life, so as to ensure adequate performance under the combined influence of these two parameters.

    - For isothermal exposure(s) used to assess thermal effects, as an independent variable, temperatures well above the highest service values are suggested.
    - Radiation exposure with a fast neutron fluence of the order of 1×10^{15} n/cm^2 is acceptable for the separate determination of radiation effects.
    - Mechanical behavior shall not be significantly compromised after exposure under either of these independent variables.

Plate materials formed using a metal matrix composite designated 1100/B4C/15p are approved for use as the neutron absorber in the TN-68 cask design. Composite materials outside of this designation are envisioned as alternative candidate materials for this application. TN could approve a candidate material only after development of appropriate qualification test data, which would ensure that the proposed plate material meets or exceeds the service requirements. In addition to qualification testing, acceptance testing shall be performed on any qualified alternative material to ensure that requirements for safety related characteristics, e.g. thermal conductivity, areal density of ^{10}B, and plate quality, are met for plates of each production run deemed to be acceptable.

The qualification tests are to be conducted at least once, for a given set of materials and manufacturing processes, to demonstrate the acceptability of the resulting product as neutron absorber plates. As specified in the CoC, criteria and methods used to establish the acceptability of the material shall be equivalent to those used for the Boralyn(TM) composite material. Many of these characteristics are in the above description of Boralyn(TM) qualification tests used for the TN-68 storage cask design.

Alternative neutron absorber plate materials may differ from the approved Boralyn(TM) designation 1100/B4C/15p. These candidate materials may have features that differ from the approved Boralyn(TM) material, as for example, alternative particle sizes or percentages of B_4C, or alternative chemical compositions of either the aluminum alloy matrix or the (boron) ceramic. They may be processed by alternative methods, e.g. cold pressing following extrusion/sintering. Adequate qualification testing requires that special consideration shall be given to the alternative features of any candidate material so as to fully understand their significance in relation to service performance and quality of the plate material produced with these features. For example, the specification for the size of particles of the approved material is typically 98 percent greater than two microns and 98 percent less than 25 microns. For any proposed use of a composite material with particle sizes outside this range, special consideration should be given to potential effects associated with particle size, e.g the degree of uniformity and the potential for neutron streaming.

The applicant takes credit for 75% of the measured value of a required minimum areal density of 36 mg/cm^2 of ^{10}B and the staff concurs that at this level of credit an effective areal density of 27 mg/cm^2 of ^{10}B will be ensured for this material. The uniformity of boron carbide particles in the plate material and the methods used to contain the B_4C particles, and to establish the effective content of ^{10}B, are regarded to be sufficient to ensure that an adequate distribution and amounts of neutron absorbing material will be present throughout the licensing period.

9.1.6 Thermal Tests

The applicant will perform thermal conductivity testing of the neutron absorber plate material to ensure the heat transfer properties are as good or better than those used in the thermal analysis. Testing may be by ASTM E1225, ASTM E1461, or an equivalent method. Tests will be performed on specimens removed from coupons and ends of finished plates. Initially, tests will be performed on one specimen per lot, with fewer tests being conducted, on an as warranted basis, after considering the results the initial tests. A lot is defined as all the plates associated coupons made from a single casting or billet. The staff believes these methods to be sufficient to ensure that the thermal performance of the neutron absorber material is adequate.

9.1.7 Cask Identification

Section 1.2.1 of the SAR states that each cask will be marked with the empty weight and an alphanumeric identifier that contains the model number and a sequential number corresponding to a specific cask (e.g., TN-68-XX). This method of identification is an acceptable means of providing a unique, permanent, and visible number to permit identification of the cask.

9.2 Maintenance Program

The TN-68 storage cask requires little maintenance over its lifetime. All safety-related functions (e.g., confinement, shielding, criticality control, etc.) are provided by passive systems and components. Typical maintenance tasks identified in the SAR include occasional recalibration of seal monitoring instrumentation, seal replacement as needed, verification of overpressure system tank pressure, and repainting. A description of the procedure for calibration of the pressure transducers/switches is provided in Section 8.3 of the SAR. The staff concludes that maintenance and inspection programs are acceptable.

9.3 Evaluation Findings

F9.1 The applicant's proposed program for pre-operational testing and initial operations of the TN-68 are described in Sections 7.1 and 9.1 and Appendix 3E of the SAR. Sections 8.3 and 9.2 of the SAR discuss the proposed maintenance program.

F9.2 SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. The safety important SSCs are identified in Section 2.3 of the SAR. The applicable standards for their design, fabrication, and testing are given under Sections 2.2 and 2.5 of the SAR.

F9.3 The applicant will examine and/or test the TN-68 to ensure the absence of defects that could significantly reduce its confinement effectiveness. Sections 7.1, 9.1, and Appendix 3E of the SAR describe this inspection and testing.

F9.4 The applicant will mark the cask with a data plate indicating its model number, unique identification number, and empty weight, as described in SAR Section 1.2.1.

F9.5 The staff concludes that the accept ınce tests and maintenance program for the TN-68 are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The acceptance tests and maintenance program are accepted as providing reasonable assurance that the cask will allow safe storage of spent fuel throughout its licensed or certified term. This finding is reached on the basis of a review that considered applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

9.4 References

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste," Title 10, Part 72.

2. ASME Boiler and Pressure Vessel Code, Section III, Division I, 1995, including 1996 addenda.

3. American Welding Society, AWS 2.4, Standard Symbols for Welding, Brazing, and Nondestructive Examination, 1986.

4. American Society for Nondestructive Testing Recommended Practice SNT-TC-1A, Personnel Qualification and Certification in Nondestructive Testing, 1984.

5. American National Standards Institute, ANSI N14.6-1993, American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials, New York, June 1993.

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

10.0 RADIATION PROTECTION EVALUATION

The NRC staff reviewed the radiation protection capabilities of the TN-68 to ensure that the cask meets regulatory dose requirements.

10.1 Radiation Protection Design Criteria and Features

10.1.1 Design Criteria

The applicant's SAR, Rev. 5, lists four major sources of radiation protection design criteria, including 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), and Regulatory Guide 8.8[1]. This is consistent with NRC guidance. The cask users are responsible for demonstrating site-specific compliance with these requirements.

10.1.2 Design Features

Sections 10.1 and 10.2.1 of the SAR describe the various radiological design features to provide radiation protection to operational personnel and members of the public. These radiation protection design features are summarized below:

- The thick walls of the TN-68 cask body provide shielding from gamma radiation.
- The cask is surrounded by a borated resin-filled layer that provides shielding from neutron radiation.
- The confinement system includes double metallic seals and an overpressure system to prevent atmospheric releases of radioactive material. The confinement system is designed to maintain confinement of radioactive materials during normal, off-normal, hypothetical accident conditions, and severe natural phenomena events.
- The cask body consists of smooth surfaces to facilitate decontamination prior to transfer to the ISFSI, to minimize the time spent decontaminating a cask, and to reduce the quantity of radioactive waste generated during decontamination.
- ALARA principles are incorporated into cask design and operating procedures to minimize the occupational exposures.

Additional radiation protection features of the TN-68 cask system include minimal maintenance and inspection requirements, location of cask monitoring instruments in an easily-accessible location, and adequate cask spacing in the ISFSI to facilitate surveillance activities.

The NRC staff evaluated the radiation protection design features and criteria for the TN-68 cask and found they provide reasonable assurance that the cask can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). In addition, all of the ALARA design considerations presented in Regulatory Guide 8.8 are addressed satisfactorily in Sections 8, 10.1.2 , and 10.1.3 of the applicant's SAR. Chapter 12 of the SAR contains TSs on the maximum allowable surface dose rates and external surface contamination levels for the cask. Sections 5, 7, and 8 of the SER discuss the staff's evaluations of the shielding capabilities, confinement features, and operating procedures, respectively. Sections

11.1 and 11.2 of the SER discuss the NRC staff's evaluation of the TN-68 cask under off-normal and accident conditions, respectively.

10.2 Occupational Exposures

Cask operating procedure descriptions that each cask user will follow for cask loading, operation, unloading, and maintenance are presented in Section 8 of the SAR. Section 10.3 of the SAR presents estimates of: (1) the time and personnel requirements for these operations, (2) the dose rates in occupied areas where these operations occur, and (3) the doses received by personnel. Operational dose rates were taken from Section 5 of the SAR. The occupational dose calculations assume no temporary shielding is used. Occupational dose estimates are given in SAR Table 10.3-1 for cask loading, transport, and emplacement and in Table 10.3-2 for cask maintenance. The estimated total dose for cask loading, transport, and emplacement is as high as 2.75 person-rem per cask. Annual maintenance doses were calculated for four maintenance activities. The highest maintenance exposure calculated was 0.197 person-rem per cask for instrument operability verification and calibration. TS 5.2.3 is provided to control surface dose rates and surface contamination limits are controlled by LCO 3.2.1 to ensure that occupational exposures are within regulatory limits.

The staff reviewed the occupational dose estimates and determined that the analysis provides reasonable assurance that use of the cask can meet the occupational exposure requirements in 10 CFR Part 20. Actual occupational exposures will depend on site-specific operating procedures and special precautions (e.g., use of temporary shielding) taken to maintain exposures ALARA. Each licensee will have an established radiation protection program required by 10 CFR Part 20 Subpart B and will also be required to demonstrate compliance with occupational limits given in 10 CFR Part 20 Subpart C and other site-specific 10 CFR Part 50 license requirements.

10.3 Public Exposures

An SAR for a dry storage cask system provides an analysis of public exposures to facilitate site-specific analyses by a cask user. The SAR for the TN-68 cask provides estimates of the public exposures assuming the distance to the controlled area boundary is 100 to 500 meters. The staff's evaluation of the applicant's analysis of public exposures during normal (SER Section 10.3.1) and hypothetical accident conditions (SER Section 10.3.2) is summarized below. Based on the following review, the NRC staff believes there is reasonable assurance that the TN-68 design, along with appropriate site characteristics, can provide the required radiation protection for members of the public.

10.3.1 Normal and Off-normal Conditions

Sections 5.1, 5.4, 7.2, and 10.2.2 of the SAR present the analysis of radiation doses during normal and off-normal operations for the TN-68 cask. The analysis shows that the confinement functions of the cask are not affected by normal and off-normal conditions. In addition, the applicant performed an analysis of a continuous, non-mechanistic release of airborne radioactive material at the tested leakage rate of the confinement system. SAR Section 5.1 presents the results of the direct-path radiation dose calculations at distances of 100 to 500 meters from the cask. Section 10.2.2 of the SAR presents the skyshine dose rates for single casks and arrays. The total dose to a member of the public at the controlled area boundary is the sum of the contributions from atmospheric releases, direct-path radiation, and skyshine. The NRC staff's review of the atmospheric release calculations is presented in SER Section 7.3

and the evaluation of the applicant's direct-path (i.e., line-of-sight) radiation dose calculations is presented in SER Section 5. The analyses were determined to be acceptable. The staff's review of the skyshine dose calculations is presented below.

Skyshine dose rates for a single bermed cask containing a design-basis fuel source were calculated by the applicant using MCNP as described in Section 5 of the SAR. The dose rates are given in SAR Table 10.2-1. The berm was assumed to be 6.1 meters high and located 20 meters from the cask centerline. Confirmatory calculations of skyshine dose rates were made with MCNP. The confirmatory calculations give a dose rate that is comparable to the applicant's calculations at a range of 100 meters and beyond. This comparison is within expected uncertainties and the NRC accepts the applicant's calculated skyshine dose rates.

The results of the applicant's site boundary analysis show that for a single cask with design-basis fuel and no berm, a minimum distance of approximately 250 meters is necessary to meet the 25 mrem/yr limit in 10 CFR 72.104(a). If a berm is placed around the cask, effectively reducing the direct radiation dose to insignificant levels, a minimum distance of about 150 meters is necessary to ensure the dose rate is below the 10 CFR 72.104(a) limits. For a typical array of 48 casks placed inside a bermed area, a minimum distance of approximately 375 meters to the nearest real person is necessary to meet the regulatory limits.

The applicant's results and staff's confirmatory analysis provide reasonable assurance that a cask user can meet the requirements of 10 CFR 72.104(a). Each cask user or general licensee must perform a site-specific analysis as required by 10 CFR 72.212(b) to demonstrate compliance with 10 CFR 72.104(a) for normal operations and anticipated occurrences. The general licensee may consider site-specific conditions, such as actual distances to the nearest real person, topography, array configurations, characteristics of stored fuel, and use of engineered features, such as berms or walls, in their analysis of public doses. The site-specific analysis must also include the doses received from other fuel cycle activities (e.g., reactor operations) in the region.

A TS that requires measured dose rates to meet established limits (see TS 5.2.3) is included in the SAR. The dose rate limits are used to identify casks that may cause the regulatory limits to be exceeded.

TS 4.3.6 has been included regarding engineered features used for radiological protection. The TS states that engineering features (e.g., berms and shield walls) used to ensure compliance with 10 CFR 72.104(a) are to be considered Important to Safety and must be evaluated to determine the applicable QA Category.

10.3.2 Accident Conditions and Natural Phenomena Events

The radiation exposures from accidents are presented by the applicant in Section 11.2 of the SAR. Accident conditions include hypothetical cask drop and tipover events, cask burial accidents, and possibly severe natural phenomena that could lead to simultaneous loss of the neutron shield and loss of one confinement barrier. The bounding dose is the sum of the direct radiation dose from loss of the neutron shield and the atmospheric dose from the loss of one confinement barrier with 100% fuel cladding failure.

Time-integrated exposures were calculated by the applicant assuming an individual is located 100 meters from the cask for 30 days. The dose rates from direct exposures to a loss of the neutron shield were calculated by scaling the normal condition dose rate at 100 meters (SAR

Table 5.1-3) to the ratio of the accident to normal dose rates at the surface (SAR Table 5.1-2). The analysis of public doses from atmospheric releases caused by loss of one confinement barrier and 100% fuel cladding failure accidents is presented in SAR Section 7.3. The accident-related doses are the sum of the time-integrated direct dose and the dose from atmospheric releases.

The NRC staff's review of the direct dose rate calculations is presented in Section 5 of the SER. The calculations, in Section 11.2.5.3 of the SAR, to estimate the dose rate at 100 meters for loss of the neutron shield were also evaluated and confirmatory analyses were performed. The time-integrated direct radiation dose at 100 meters after the assumed loss of neutron shielding was calculated to be about 504 mrem, assuming an individual is present for an entire 30-day period. The staff's review of the doses from loss of one confinement barrier and 100% fuel failure is presented in SER Section 7. The total effective dose equivalent (TEDE) from this event was calculated to be 92.4 mrem at 100 meters from the cask. The total dose of about 598 mrem, including 2 mrem background dose from the rest of the ISFSI, was found to be well below the 5 rem limit set forth in 10 CFR 72.106(b). The staff concludes there is reasonable assurance that the combined doses from direct radiation and atmospheric releases from bounding design-basis accidents and natural phenomena will be below the 5 rem regulatory limit specified in 10 CFR 72.106(b).

10.4 ALARA

The TN-68 shielding design incorporates a number of features to maintain radiation exposures ALARA. Operational ALARA policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20. The staff evaluated the ALARA assessment of the TN-68 and found it to be acceptable. TSs are provided that include surface dose rates (see TS 5.2.3) and surface contamination limits (see LCO 3.2.1) to ensure that occupational exposures are maintained ALARA.

10.5 Evaluation Findings

F10.1 The TN-68 provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 72.106.

F10.2 The occupational radiation exposure estimates are within the limits of 10 CFR Part 20 and meet the objective of maintaining exposures ALARA.

F10.3 The staff concludes that the design of the radiation protection system of the TN-68 is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the TN-68 will allow safe storage of spent fuel as required by the regulations. This finding is reached on the basis of a review that considered the specifications in the SAR, the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

10.6 References

1. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable," Revision 3, U.S. Nuclear Regulatory Commission, June 1978.

11.0 ACCIDENT ANALYSES

The purpose of the review of the accident analysis is to evaluate the applicant's identification and analysis of hazards, as well as the summary analysis of system responses to both off-normal and accident or design-basis events. This ensures that the applicant has conducted thorough accident analyses, as reflected by the following factors:

- Identified all credible accidents
- Provided complete information in the SAR
- Analyzed the safety performance of the cask system in each review area
- Fulfilled all applicable regulatory requirements

The conclusions in this SER section are based on information provided in TN-68 SAR Revision 5.

11.1 Off-Normal Events

Section 11.1 of the SAR examines the causes, radiological consequences, and corrective actions for the off-normal events described below. The SAR analysis demonstrated that the confinement function of the TN-68 is not affected by off-normal conditions. However, because the cask lid seals are not demonstrated to be leak tight as defined in ANSI N14.5-1997[1], SAR Section 7 presented an analysis of the release of radioactive materials at the tested seal leakage rate as corrected for off-normal conditions. The radiological doses from this continuous atmospheric release are evaluated in Section 7.4 of the SER.

The staff reviewed the off-normal event analysis, performed confirmatory calculations, and found the estimated dose consequences were within the allowable limits. Therefore, staff has reasonable assurance that the dose to any individual beyond the controlled area boundary will not exceed the limits in 10 CFR 72.104(a)[2] during off-normal conditions and anticipated occurrences.

11.1.1 Loss of Electric Power

The applicant analyzed a loss of electric power as an off-normal event in SAR Section 11.1.1. Electric power provides area lighting and power to the OMS instrumentation. Neither area lighting nor the OMS instrumentation are important to safety. Because loss of electric power has no effect on confinement boundaries, there would be no radiological consequences from this event.

11.1.2 Cask Seal Leakage or Leakage of the OMS

SAR Section 11.1.2 presents analysis for cask seal leakage or leakage of the OMS. If the OMS is functioning, leakage from the cask is not expected as discussed in SER Section 7.2. As a bounding off-normal case, the OMS pressurization function is assumed to have failed and leakage of radioactive material from the cask is assumed for a source term and leak rate calculated using the methodology discussed in SER Section 7.3 for off-normal conditions. Radiological dose estimates from off-normal conditions presented in Section 7.3 of the SAR demonstrate that the TN-68 design meets the requirements of 10 CFR 72.104(a).

11.1.3 Overpressure Tank Needs Refilling

Slow leakage across the inner or outer seals at less than the allowable leakage rate will result in the need to refill the overpressure tank. An analysis of this off-normal condition is presented in SAR Section 11.1.3. The applicant also performed calculations in the SAR Section 7.1.5 that demonstrated that it would take about 10 – 12 years at the tested leak rate to reach the OMS alarm setpoint. The staff's review of these calculations is in SER Section 7.2. Based on this review, the staff concludes that there is reasonable assurance that there should be no radiological dose consequences because the OMS alarm setpoint is selected such that there is sufficient time to repressurize the overpressure tank before any leakage would occur from the cask cavity. In addition, SAR Section 11.1.3 states that maintenance to repressurize the OMS will be performed as a scheduled maintenance activity. The staff concluded that the TN-68 meets radiological dose limits for off-normal OMS pressure conditions.

11.2 Design-Basis Accidents and Natural Phenomena Events

Section 11.2 of the SAR examines the causes, radiological consequences, and corrective actions for the identified design-basis accidents and natural phenomena events. The SAR demonstrated that the TN-68 would reasonably maintain its confinement function during and after design-basis accidents. However, SAR Section 11 evaluates the radiological doses from a combination of unlikely events, including events that result in loss of neutron shielding, loss of one confinement barrier, and simultaneous loss of neutron shielding and one confinement barrier. The applicant determined that the radiological dose at 100 meters would not exceed the dose limits specified in 10 CFR 72.106(b)[3].

The staff reviewed the design-basis accident analyses, performed confirmatory analyses, and found the estimated dose consequences to be within the allowable limits. Therefore, the staff has reasonable assurance that the dose to any individual beyond the controlled area boundary will not exceed the limits in 10 CFR 72.106(b) for credible hypothetical accident conditions. Sections 5, 7, and 10 of the SER provide further evaluations of the radiological doses during accident conditions.

11.2.1 Earthquake

11.2.1.1 Cause of Earthquake

An earthquake at the ISFSI site is postulated.

11.2.1.2 Consequences of Earthquake

The applicant performed a seismic event analysis to determine the effects of a design-basis earthquake on the TN-68 storage cask. This analysis demonstrated the stability of the TN-68 under application of vertical and horizontal seismic loading conditions. SAR Section 2.2.3 qualified the cask for an applied acceleration up to 0.26 g horizontal and 0.17 g vertical loading conditions. The TN-68 will not tipover or slide under the equivalent seismic loading conditions. The staff concludes that the cask would maintain confinement under these applied loading conditions and that no radiological or safety consequences result from this event.

11.2.2 Extreme Winds and Tornado Missiles

11.2.2.1 Causes of Extreme Wind and Tornado Missiles

The TN-68 cask will be placed on an unsheltered concrete pad at an ISFSI and will be subject to extreme weather conditions that could include extreme winds from a tornado. Extreme winds may also generate missiles that could strike the cask.

11.2.2.2 Consequences of Extreme Wind and Tornado Missiles

High-velocity winds from passing tornadoes will exert an external pressure load on the cask and could also generate large missiles that have the potential for striking and damaging the cask. The potential effects include cask tipover or penetration of the cask confinement boundary.

The analysis of cask stability under extreme wind loading conditions was presented in SAR Section 2.2.1. Cask stability was evaluated for a design-basis wind velocity of 360 mph and a pressure drop of 3 psi. The cask was shown to not tipover or slide as a result of the 360 mph wind. The external pressure drop of 3 psi, when combined with other internal pressure loads, was shown to be small when compared to the 100 psi design internal pressure of the cask.

SAR Section 2.2.1 also evaluates the stability of the cask and the potential damage to cask structures when subjected to impacts from various missiles. These analyses showed that the loaded TN-68 cask may slide a short distance as a result of a missile impact. The applicant determined that the cask will remain upright under simultaneous tornado wind and tornado missile loadings and that tornado missiles will not breach the cask confinement boundary. Staff review of tornado missile impacts is in SER Section 3.4.4.

A tornado missile may damage the OMS and the neutron shield. To determine the bounding radiological dose consequences from this accident, the applicant combined the doses from the loss of neutron shielding, in SAR Section 11.2.5.3, with the TEDE from the loss of one confinement barrier with 100% fuel cladding failure, in SAR Section 11.2.9.3. The estimated combined dose from this accident is about 598 mrem, which is below the limits in 10 CFR 72.106(b).

11.2.3 Flood

11.2.3.1 Causes of Floods

A flood at an ISFSI caused by external events such as unusually high water from a river, dam break, seismic event, tsunami, and severe weather (e.g., hurricanes) is postulated.

11.2.3.2 Consequences of Floods

The analysis of flood effects on the TN-68 is presented in SAR Section 2.2.2. The cask was evaluated for a 57 ft static head of water (25 psi). The cask confinement boundary would not be compromised for static heads less than 57 ft. The cask was also evaluated for a water drag force of 45,290 lbs. This is equivalent to a stream of water flowing past the cask at 22.1 ft/sec. The analysis demonstrates that the cask would not tipover or slide at this water velocity. However, a flood could damage the OMS or deposit debris around the cask. The bounding radiological dose consequences from this accident are the combined dose from the loss of neutron shielding, in SAR Section 11.2.5.3, with the TEDE from the loss of one confinement barrier with 100% fuel cladding failure, in SAR Section 11.2.9.3. The estimated combined dose from this accident is about 598 mrem, which is below the limits in 10 CFR 72.106(b). The impacts of flood debris are bounded by the cask burial analysis in SAR Section 11.2.10.

11.2.4 Explosion

11.2.4.1 Causes of Explosions

An explosion caused by combustion of the cask transporter fuel is credible. Explosions involving combustible materials shipped to reactor sites and on transportation links near nuclear power plant sites are also possible.

11.2.4.2 Consequences of Explosions

The external pressure wave generated in a credible explosion accident is on the order of a few psig. This is bounded by the design-basis external pressure of 25 psig. The structural evaluation in Section 3 of the TN-68 SAR demonstrated that the stresses on confinement boundary components from the design-basis external pressure are within allowable levels. The analysis demonstrates there will be no effect on the integrity of the confinement boundary as a result of a credible explosion event.

11.2.5 Fire

11.2.5.1 Causes of Fire

A rupture of the transporter vehicle fuel tank and subsequent ignition of a 200 gallon pool of spilled fuel is postulated.

11.2.5.2 Consequences of Fire

As described in SAR Section 4.5.1, a bounding, hypothetical fire is assumed that engulfs the cask and burns for 15 minutes. After the fire burns out, the post-fire thermal transient is evaluated. The thermal analysis in Section 4.5 of the SAR demonstrated that this fire would not compromise the integrity of the TN-68 confinement systems, that no melting of cask components would occur, and that the fuel cladding remains below its maximum short-term temperature limit. The applicant's internal pressurization analysis in SAR Section 7.3.2.2 demonstrates that the cask cavity pressure remains below the design pressure of 100 psig. Based on this analysis, the applicant concluded there would be no release of radioactive material from the cask. Staff review of the temperature and pressure analysis in SER Section 4 concludes that the fire accident component temperatures and cask internal pressure were within limits and, therefore, would not compromise the integrity of the cask.

However, the applicant concluded that the neutron shield would offgas during the hypothetical fire. As a result, a shielding analysis in Section 5 of the SAR calculated the radiation dose rate assuming the neutron shield resin is removed. Staff review of the shielding calculations is presented in Section 5 of the SER. The dose at the site boundary was calculated to be about 506 mrem, assuming the off-site receptor is located 100 meters away from a cask continuously for a 30-day period (no berm was assumed). The applicant's estimate of the site boundary accident dose was an approximation. This approximation scaled the differences in surface dose rates for the normal and accident case and used that value to multiply the normal dose rate at 100 meters to derive an accident dose rate. The staff concludes that this simplified approximation method was acceptable because the resulting dose was only a small fraction of the 5 rem limit of 10 CFR 72.106(b).

11.2.6 Inadvertent Loading of A Newly-Discharged Fuel Assembly

11.2.6.1 Causes of Loading a Newly-Discharged Fuel Assembly

An operator error or failure of administrative controls governing fuel handling operations is postulated.

11.2.6.2 Consequences of Loading A Newly-Discharged Fuel Assembly

The SAR evaluated the loading of a fuel assembly with a decay heat load greater than the design-basis 0.312 kW/assembly. As discussed below, the loading with spent fuel with a higher than design-basis decay heat load is not a credible accident.

The TN-68 SAR specifies the parameters for spent fuel assemblies allowed to be stored in the cask. In addition, TS 2.1 specifies the allowable fuel parameters. SAR Table 8.1-1 states that (1) preselected fuel assemblies will be loaded into the cask, (2) procedures will be developed by the cask users to ensure that the fuel to be loaded meets the fuel specifications, and (3) the fuel assembly identities are to be verified after they are loaded into the cask, including, as stated in SAR Section 11.2.6.2, a final verification of the fuel loaded into the cask and comparison with fuel management records. Fuel loading and measures to ensure that the TS requirements are met will be conducted under the cask user's quality assurance program (QAP). This provides the staff with reasonable assurance that this condition will be detected and appropriate corrective actions will be taken prior to sealing 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 Causes of Improper Cask Loading

An operator error or failure of administrative controls governing fuel handling operations is postulated.

11.2.7.2 Consequences of Improper Cask Loading

The SAR evaluated the loading of a fuel assembly with an initial enrichment of 5 wt% U-235, which is greater than the design-basis initial enrichment of 3.7 wt% U-235. All fuel is modeled as fresh fuel. In Section 6 of the SAR, the applicant demonstrated that the cask remains subcritical under these conditions.

As discussed in SER Section 11.2.6.2, there are sufficient controls to detect and correct this loading error prior to sealing the cask. Therefore, there are no consequences from this event and no adverse effects on the cask system.

11.2.8 Hypothetical Cask Drop and Tipping Accidents

11.2.8.1 Causes of Cask Drop and Tipover Accidents

Although a handling accident event is unlikely, a cask drop from the handling height limit during transport is regarded as a credible event. The analysis of the TN-68 has shown that the cask does not tipover as a result of severe natural phenomena, such as earthquakes, tornadoes, tornado missiles, and floods. However, a cask tipover is evaluated as a bounding event to demonstrate the defense-in-depth of the design.

11.2.8.2 Consequences of Cask Drop and Tipover Accidents

Cask drop and tipping accidents are analyzed in Section 11.2.8 of the TN-68 SAR. The effects of drop (18-inch) and tipping accidents on the cask structures were evaluated in Section 3 of the SAR. Staff review of the structural analyses of these hypothetical accidents are located in Section 3 of this SER.

Sections 2.2, 3, and 11 of the TN-68 SAR qualify the cask for a bottom end drop deceleration of 60 g and side deceleration of 65 g, to simulate bounding loads for the end drop and tipover accidents, respectively. The cask was shown to maintain confinement after these deceleration conditions were applied. TS 5.2.2 is provided to control cask lift height or take other actions to ensure that decelerations are maintained below these levels. TS 3.1.6 is provided to ensure that the ambient temperature and cask external surface temperature are above -20°F to preclude brittle fracture concerns. These controls assure that the consequences of cask drop and tipover accidents are limited to those analyzed.

A drop accident could damage the OMS and the neutron shield. To determine the bounding radiological dose consequences from this accident, the applicant combined the doses from the loss of neutron shielding, in SAR Section 11.2.5.3, with the TEDE from the loss of one confinement barrier with 100% fuel cladding failure, in SAR Section 11.2.9.3. The estimated combined dose from this accident is about 598 mrem, which is below the limits in 10 CFR 72.106(b).

11.2.9 Loss of Confinement Barrier

11.2.9.1 Causes of a Loss of Confinement Barrier

A loss of OMS integrity and leakage from the confinement is postulated.

11.2.9.2 Consequences of a Loss of Confinement Barrier

The dose calculations for a loss of confinement are presented in Section 7.3 of the SAR. The cask is assumed to leak for 30 days at the tested leak rate (1×10^{-5} ref cm^3/sec) as adjusted for confinement temperatures and pressures assuming 100% fuel cladding failure and internal temperatures at the maximum post-fire level. Staff review of this analysis is presented in SER Section 7.4. As summarized in SER Table 7-2 for an assumed minimum site boundary of 100 meters, the calculated TEDE was 92 mrem and the maximum organ (bone surface) dose was 404 mrem. These calculated doses are well within the allowable limits and provide reasonable assurance that the doses at the controlled area boundary will not exceed the maximum dose criteria in 10 CFR 72.106(b).

11.2.10 Buried Cask

11.2.10.1 Cause of Buried Cask

The SAR analyzed the effects of cask burial that may result from an earthquake or other natural phenomena that could lead to burial of the cask under man-made or earthen material.

11.2.10.2 Consequences of a Buried Cask

Thermal analyses were performed in Section 4.5.2 of the SAR to evaluate cask temperatures assuming it is completely buried in a medium that interferes with natural convection cooling and unrestricted radiative heat transfer to the environment. The results indicated that the neutron shield would begin to degrade after 12 hours and that cask seals would reach their long-term maximum temperature limit after about 64 hours. Accordingly, actions to retrieve the cask should begin as soon as possible to prevent seal failure. The staff has reasonable assurance that a minimum of 64 hours before seal temperature limits are met provides sufficient time to implement corrective actions to prevent seal failure.

The dose consequences from this event would be represented by loss of the neutron shield combined with loss of one confinement barrier. To determine the bounding radiological dose consequences from this accident, the applicant combined the doses from the loss of neutron shielding, in SAR Section 11.2.5.3, with the TEDE from the loss of one confinement barrier with 100% fuel cladding failure, in SAR Section 11.2.9.3. The estimated combined dose from this accident is about 598 mrem, which is the same as the bounding radiological dose from the extreme wind and cask drop and tipover accidents analyzed in Sections 11.2.2 and 11.2.8 of the SER. The resulting doses are well below the accident dose limits in 10 CFR 72.106(b).

11.2.11 Latent Seal Failure

11.2.11.1 Causes of Latent Seal Failure

Although extremely unlikely, it is possible for a seal to leak at a rate greater than the tested leak rate during the 20-year storage period. For postulated seal leakage rates greater than the tested rate, but not gross leakage, there could be a lag time before OMS pressure decays to 3.0 atm abs and indicates a low pressure condition. This degraded seal leakage is considered a "latent" condition and should be presumed to exist concurrently with other design-basis events. If the outer seal has the latent failure or the OMS leaks, the inner seal still functions to provide confinement and there will be no release from the cask cavity. Of interest is a postulated latent failure of the inner seal concurrent with a postulated loss of OMS integrity from a design-basis accident.

11.2.11.2 Consequences of Latent Seal Failure

The applicant provided the results of sensitivity studies in SAR Section 7.3.3 to determine (1) the delay time from the onset of the latent condition to the point where the OMS would indicate system leakage, and (2) the dose consequences if an accident occurred concurrent with the latent condition. For a leak rate of 1×10^{-3} ref cm^3/sec (100 times the tested rate), a maximum delay time to indication of a degraded seal was 16 days. At this leak rate, the applicant's dose assessment concluded that the dose limits of 10 CFR 72.106(b) would be exceeded after the leak condition existed for 16 days. The staff concludes that the analysis of the latent seal failure was acceptable as discussed in Section 7.5 of the SER.

11.3 Criticality

Nuclear criticality evaluations are presented in Section 6 of the SAR. Confinement is maintained during credible hypothetical accidents and natural phenomena events, which will prevent fresh water from entering the cask cavity. However, for loading and unloading operations, the cask has been demonstrated to remain subcritical while flooded with fresh water at various water densities and levels. The staff concludes that subcriticality will be maintained during normal operations and after credible accidents and natural phenomena events that could occur during dry storage.

As discussed in Section 6 of the SER, the applicant has shown that the irradiated fuel remains subcritical ($k_{eff} < 0.95$) under all credible normal, off-normal, and postulated accident conditions. The design-basis off-normal and credible accident conditions do not adversely affect the design features important to criticality safety. Based on the assessment provided in Section 6 of the SER, the staff concludes that the TN-68 meets the "double-contingency" requirements of 10 CFR 72.124(a).

11.4 Post-Accident Recovery

Section 11.2 of the SAR discusses corrective actions for each accident identified in Section 11.2. The SAR did not identify a design-basis accident that would affect the confinement boundary or significantly damage the cask system at a level that could result in undue risk to public health and safety.

The staff reviewed the design-basis accident analyses with respect to post-accident recovery and found them to be acceptable. The staff has reasonable assurance that the site licensee can recover the TN-68 cask from the analyzed design-basis accidents and that the generic corrective actions outlined in the SAR are appropriate to protect public health and safety.

11.5 Instrumentation

Because the TN-68 is a passive storage system, no instrumentation and control systems are required to remain operational under accident conditions. The SAR demonstrated that the confinement boundary integrity would be maintained during normal, off-normal, and design-basis accident and natural phenomena conditions. However, cask seal monitoring provided by the OMS may not be functional following an accident or severe natural event. The applicant evaluated the radiological consequences for seal leakage without the OMS function and demonstrated that estimated doses would only be a fraction of the 10 CFR 72.106 (b) limits. The applicant demonstrated that doses could be maintained within limits in the extremely unlikely event of an undetected, or latent, seal failure and a concurrent accident that breaches the OMS boundary. Post-accident recovery actions include verification and/or restoration of proper OMS function to ensure that technical specification surveillance requirements are met. The staff concludes that no TN-68 instrumentation is required to remain operational under accident conditions.

11.6 Evaluation Findings

F11.1 The SSCs of the TN-68 storage cask are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that may occur.

F11.2 The spacing of casks, discussed in Section 4.4.1.4 of the SER and included as TS 4.2.1, will ensure accessibility of the equipment and services required for emergency response.

F11.3 Table 12-1 of this SER lists the TS for the TN-68. These TS are further discussed in Section 12 of the SER.

F11.4 The applicant has evaluated the TN-68 to demonstrate that it will reasonably maintain confinement of radioactive material under credible accident conditions.

F11.5 An accident or natural phenomena event will not preclude the safe recovery of the TN-68 spent fuel cask.

F11.6 The spent fuel will be maintained in a subcritical condition under accident conditions.

F11.7 Neither off-normal nor accident conditions will result in a dose to an individual outside the controlled area that exceeds the limits of 10 CFR 72.104(a) or 72.106(b), respectively.

F11.8 No instrumentation or control systems are required to remain operational under accident conditions.

F11.9 The staff concludes that the accident design criteria for the TN-68 are in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the cask adequately demonstrates that it will provide for safe storage of spent fuel during credible accident situations. This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

11.7 References

1. American National Standards Institute (ANSI), ANSI N14.5-1997, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," New York, New York, February 1997.

2. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Waste," Title 10, Part 72.104, as revised Federal Register, Vol. 63, No. 197, p. 54559, October 1998.

3. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Waste," Title 10, Part 72.106, as revised Federal Register, Vol. 63, No. 197, p. 54559, October 1998.

12.0 CONDITIONS FOR CASK USE - TECHNICAL SPECIFICATIONS

The conditions for cask use are reviewed to ensure the applicant has fully evaluated the TS and that the SER incorporates any additional operating controls and limits that the staff determines are necessary.

12.1 Conditions for Use

The conditions for use of the TN-68 are fully defined in the CoC and the TS which are appended to it.

12.2 Technical Specifications

SER Table 12-1 lists the TS for the TN-68 dry storage cask system. The staff has appended these TS to the CoC for the TN-68.

12.3 Evaluation Findings

F12.1 Table 12-1 of the SER lists the TS for the TN-68. These TS are further discussed in Section 12 of the SAR and are part of the CoC.

F12.2 The staff concludes that the conditions for use of the TN-68 identify necessary TS to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The TS provide reasonable assurance that the cask will provide for safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

Table 12-1 TN-68 Technical Specifications

Number	Technical Specification
1.0	USE AND APPLICATION
1.1	Definitions
1.2	Logical Connectors
1.3	Completion Times
1.4	Frequency
2.0	FUNCTIONAL AND OPERATING LIMITS
2.1	Functional and Operational Limits
2.1.1	Fuel to be Stored in the TN-68 Cask
2.2	Functional and Operating Limits Violations
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
3.0	SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	CASK INTEGRITY
3.1.1	Cask Cavity Vacuum Drying
3.1.2	Cask Helium Backfill Pressure
3.1.3	Cask Helium Leak Rate
3.1.4	Combined Helium Leak Rate
3.1.5	Cask Interseal Pressure
3.1.6	Cask Minimum Lifting Temperature
3.2	CASK RADIATION PROTECTION
3.2.1	Cask Surface Contamination
4.0	DESIGN FEATURES
4.1	STORAGE CASK
4.1.1	Criticality
4.1.2	Structural Performance
4.1.3	Codes and Standards
4.1.4	Helium Purity
4.2	STORAGE PAD
4.2.1	Storage Locations for Casks
4.3	ISFSI SPECIFIC PARAMETERS AND ANALYSIS

Table 12-1 TN-68 Technical Specifications

Number	Technical Specification
5.0	ADMINISTRATIVE CONTROLS
5.1	TRAINING MODULE
5.2	PROGRAMS
5.2.1	Cask Sliding Evaluation
5.2.2	Cask Transport Evaluation Program
5.2.3	Cask Surface Dose Rate Evaluation Program

13.0 QUALITY ASSURANCE

Part 72 of Title 10 of the Code of Federal Regulations (10 CFR), provides for "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."[1] Subpart G of 10 CFR Part 72 describes Quality Assurance (QA) requirements applying to ISFSIs.

The SAR section on Quality Assurance (QA) states that all quality related activities will be controlled under an NRC approved quality assurance program meeting the requirements of 10 CFR Part 72. The TN QA Program was reviewed and approved by staff under separate correspondence.

13.1 References

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste," Title 10, Part 72.

14.0 DECOMMISSIONING

The purpose of the review of the conceptual decommissioning plan for the TN-68 dry storage cask system is to ensure that it provides reasonable assurance that the cask owner can conduct decontamination and decommissioning in a manner that adequately protects the health and safety of the public. Nothing in this review considers, or involves the review of, ultimate disposal of spent nuclear fuel.

14.1 Decommissioning Considerations

The conceptual decommissioning plan for the TN-68 is provided in Section 14 of the SAR. TN presents two decommissioning options. In Option 1, the TN-68, including the spent fuel in storage, is shipped to either a monitored retrievable storage system or geological repository for final disposition. In Option 2, the spent fuel is removed from the cask and shipped in an NRC approved cask. Table 14.1-2 of the SAR provides the activity concentrations of the major radiation sources in the cask which TN has determined would exist after 40 years of irradiation by 68 design-basis BWR fuel assemblies stored in the TN-68 system and 30 days decay. The material activation results presented in Table 14.1-2 confirm that total system activation is low for all components. While the applicant has not kept the methods in this section current with the methods used in Section 5 of the SAR, the staff accepts this analysis because of the large margins in the results and the conceptual nature of the decommissioning plan.

TN determined that the TN-68 cask could be decommissioned using standard industry practices. Activated steel components can be decontaminated using existing mechanical or chemical methods.

14.2 Evaluation Findings

F14.1 The TN-68 system design includes adequate provisions for decontamination and decommissioning. As discussed in Section 14 of the SAR, these provisions include facilitating decontamination of the TN-68, if needed; storing the remaining components, if no waste facility is expected to be available; and disposing of any remaining low-level radioactive waste.

F14.2 Section 14 of the SAR also presents information concerning the proposed practices and procedures for decontaminating the cask and disposing of residual radioactive materials after all spent fuel has been removed. This information provides reasonable assurance that the applicant will conduct decontamination and decommissioning in a manner that adequately protects public health and safety.

F14.3 The staff concludes that the decommissioning considerations for the TN-68 are in compliance with 10 CFR Part 72. This evaluation provides reasonable assurance that the TN-68 will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

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