Safety Evaluation Report

Model No. NAC-STC
Certificate of Compliance No. 9235
Revision No. 4

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SUMMARY

By application dated November 8, 2000, as supplemented, NAC International (NAC) requested an amendment to Certificate of Compliance (CoC) No. 9235, for the Model No. NAC-STC package. NAC requested that the CoC be amended to (1) include a "-85" designation, and (2) authorize shipment of canistered Connecticut Yankee Class fuel and Greater Than Class C (GTCC) waste. Additionally, NAC revised the package drawings to reflect improved cask and component fabrication procedures, and to incorporate miscellaneous administrative and editorial changes. NAC evaluated the package design to the requirements of 10 CFR Part 71 that became effective on April 1, 1996. NAC provided updated structural, thermal, shielding, confinement, and criticality analyses for the package, as well as updated operating procedures and an updated maintenance program.

The staff concludes that the changes requested by NAC will not affect the package's ability to meet the requirements of 10 CFR Part 71. Therefore, pursuant to 10 CFR Part 71, CoC No. 9235 is amended.

REFERENCES

NAC International, application dated November 8, 2000.

Supplements dated December 14, 2001, December 28, 2001, February 21, 2002, March 22, 2002, May 31, 2002, June 13, 2002, July 8, 2002, and August 23, 2002.

1.0 GENERAL INFORMATION

The NAC-STC system with the Connecticut Yankee (CY) basket is designed to transport up to: (1) 26 zircaloy clad spent fuel assemblies enriched up to 3.93 wt. percent ^{235}U, (2) 26 stainless steel clad spent fuel assemblies enriched up to 4.03 wt. percent ^{235}U, or 3) 24 zircaloy clad spent fuel assemblies enriched up to 4.61 wt. percent ^{235}U. The 24- and 26-assembly baskets are identical, except that 2 guide tubes of the 26-assembly basket are mechanically blocked to create the 24-assembly basket. In addition, up to four Connecticut Yankee reconfigured fuel assemblies (RFAs) or damaged fuel canisters (DFCs) are permitted, when placed in the four oversized corner guide tubes.

1.1 Drawings

The applicant submitted 40 revised drawings. Four are for the NAC-STC quarter scale model. The remaining 32 are for the CY-STC system components, and associated components and for the Yankee-MPC system. The revision reflected minor changes to improve cask and component fabrication procedures based on NAC's previous

fabrication activities. Also, some administrative and editorial corrections are incorporated into the drawings.

2.0 STRUCTURAL EVALUATION

The amended NAC-STC system, which weighs up to 260,000 lbs, has components and operating features similar to the previously approved NAC-STC for transporting the CY spent fuel assemblies and GTCC waste. The impact limiters and a number of major components of the system are reconfigured although the same shipping cask of the previously approved system continues to be used. In the following evaluation, after reviewing the design features unique to the amended NAC-STC, the staff focused on the determination of impact loads resulting from the balsa impact limiter option and the corresponding structural performance of the system components under load combination effects of mechanical, pressure, and thermal conditions.

2.1 Design Features

Section 1.1 of the application provides a general description of the NAC-STC system, which also identifies the design features unique to the amended NAC-STC. Compared to the previously approved NAC-STC, the amended system is configured with: (1) the same shipping cask but for an increased package weight from 250,000 lbs to 260,000 lbs, (2) an increased length from 122.5 inches to 151.75 inches, (3) a fuel basket with varied cutout size and pattern for the support disk, (4) a new Greater Than Class C (CY-GTCC) waste basket assembly to accommodate 24 containers, and (5) two balsa impact limiters constructed primarily of balsa, except for the corner redwood wedge blocks, for the packages weighing up to 260,000 lbs. A Connecticut Yankee reconfigured fuel assembly (CY-RFA) and damaged fuel can (CY-DFC) are also used to confine failed fuel rods and to hold a complete intact or damaged fuel assembly, respectively.

The fuel basket assembly is laterally supported by 28 equally spaced support disks, which, in turn, are axially retained by split spacers aligned on six tie rods. The support disk has 26 cutouts to accommodate fuel tubes. Four corner locations with oversized cutouts are available for loading CY-RFAs and CY-DFCs.

The CY-RFA consists of a 10 x 10 array of stainless steel tubes attached to the upper and lower end fittings that are similar to those used on a standard fuel assembly. It measures 8.9 inch square by 141.5 inch long to allow its placement only in the four basket corner locations with oversized cutouts in the support disk. Four ½-inch thick grid plates provide lateral support for the tubes, and four 2 x 2 x 3/16 angles, one at each corner, are used to provide additional strength to the assembly.

The 8.9 inch square by 141.5 inch long CY-DFC consists of an 18-gage stainless steel shell body, a bottom weldment and a top closure assembly. It is designed to hold a Connecticut Yankee damaged fuel assembly, Lattice, or Failed Rod Storage Canister by providing confinement function yet allowing release of gaseous products and liquids.

The stainless steel CY-GTCC waste basket assembly consists of two major parts: (1) the shield shell weldment constructed with 1.75-inch-thick plates, which form the wall of a cylinder 141.5 inches long and 64 inches in inside diameter, and (2) the tube array weldment made up with 24 identical tubes each with an 8.74-inch square inside dimensions and a 0.375-thick wall. The shield shell is held inside the TSC by twelve 1-1/4-inch thick plates or bars equally spaced and welded along the full length of the external surface of the shell wall. The 24 tubes, at a maximum of six in a row, are stacked and held together by connecting adjacent tubes with welded bars and angles for all twelve perimeter tubes. The tube array weldment rests on the tube corners in contact with the shield shell when in a horizontal position. The tube array is prevented from rotation by the alignment tabs in the inside of the shield shell weldment

2.2 Free Drop Tests Evaluation of the Package

The applicant used the 30-ft drop tests of a 1/4-scale model to demonstrate that: (1) the impact limiter crush depths are limited to prevent the cask body from direct contact with the impact surface, (2) the cask rigid-body decelerations are bounded by those used in the package design analysis, and (3) the impact limiters remain attached to the cask body and in position after a drop event. Additionally, the drop tests were used to benchmark a finite element analysis approach from which crush depths of the impact limiters and rigid-body decelerations of the cask body can be conservatively calculated for the amended NAC-STC system.

2.2.1 1/4-Scale Cask Model Drop Tests

The applicant modified the ends and weight of the 1/4-scale cask body used for the previously approved NAC-STC for the 30-ft drop tests of the cask model equipped with two 1/4-scale balsa impact limiters. Section 2.10.12 of the application describes the top end-drop, top corner-drop and side-drop tests performed at the Sandia National Laboratory. Included also in the description are the instrumentation and data reduction programs for recording cask impact response time histories, measuring impact limiter crush depths, and low-pass filtering raw data to obtain rigid-body decelerations of the cask body. The staff reviewed the program implementation and finds that test results, such as peak rigid-body decelerations and corresponding response pulse durations and shapes, are adequately reduced. The tests confirmed the capabilities of the impact limiters. They also provided the needed data for benchmarking a finite element analysis approach to calculating cask impact response as discussed below.

2.2.2 LS-DYNA Analysis of the NAC-STC 1/4-Scale Model

Using the general purpose, explicit, finite element code LS-DYNA, which is commercially available, the applicant developed a finite element analysis approach for calculating the drop test response of the 1/4-scale NAC-STC cask equipped with two balsa impact limiters. Section 2.10.12.7 of the application presents details of the finite element model constructed primarily of 8-node bricks and 4-node shells.

Recognizing the primary objective of calculating rigid-body responses of the cask body, the applicant used a single-shell representation, but with adjusted elastic modulus, to simulate the cross sectional properties of the cask body and its steel-lead-steel multi-

wall construction. The applicant performed dynamic crush tests of wood specimens to obtain stress-strain curves at varied strain rates to model the redwood and balsa properties with the LS-DYNA Modified_Crushable_Foam material option. Other standard LS-DYNA options used included the Piecewise_Linear_Plasticity material type for large deformations of steel shell and gussets of the impact limiter, Surface_to_Surface contact for interfaces between the cask body and impact limiters, and the Rigidwall_Geometric_Flat option to represent unyielding impact surface. An initial velocity of 527.4 in/sec was applied to the entire model to represent the 30-ft drop.

The applicant did not perform quasi-static tests of impact limiter models to obtain load-deflection curves. In the December 14, 2001, response to the staff request for additional information, however, the applicant performed numerical simulations to calculate force-deflection curves of the 1/4-scale impact limiter subject to the end, corner, and side drops. Since the curves are characteristic of quasi-static impact limiter tests and explicit load-deflection curves are not part of the LS-DYNA finite element modeling of the impact limiter, the staff agrees that quasi-static tests of the balsa impact limiter need not be performed.

Figures 2.10.12-4, -8, and -13 of the application compare the calculated with the tested rigid-body deceleration time histories for the 30-ft side-, top corner-, and top end-drop tests, respectively. The calculated response pulse durations and shapes correlate well with those obtained by the tests. Furthermore, since the calculated peak decelerations also bound the corresponding test results, as shown in the table below, the staff concludes that the LS-DYNA finite element analysis approach is adequately benchmarked for application to the NAC-STC system with the balsa impact limiters.

Calculated and Tested Peak Decelerations - 1/4-Scale Model, 30-Ft Drop				
Drop Orientation	Drop Test (g)	Drop Test (g)	Calculated (g)	Calculated (g)
	Top	Bottom	Top	Bottom
Top Corner	126	---	137	---
Top End	122	---	128	---
Side	150	164	184	179

2.2.3 LS-DYNA Analysis of the NAC-STC with Balsa Impact Limiters

Following the analytical modeling approach benchmarked above, the applicant constructed a finite element analysis model of the amended NAC-STC to calculate impact limiter crush depths and cask rigid-body decelerations. Section 2.6.7.4.2 of the application presents the LS-DYNA model and analysis results for the package. The Modified_Crushable_Foam material type continued to be used to model the balsa impact limiters to allow for the input of stress-strain curves for strain rate effects. To account for crush strength fabrication tolerances and temperature effects, stress values

of the stress-strain curves for the hot (200°F) and cold (-40°F) conditions were further adjusted by the factors of 0.90 and 1.10, respectively.

Figures 2.6.7.4.2-5 through -7 of the application present calculated responses for the 1-ft free drop tests for the side, c.g.-over-corner, and end drops. The peak cask rigid-body deceleration of 16.5 g occurs in the side-drop.

Figures 2.6.7.4.2-8 through -10 present the calculated deceleration responses for the 30-ft drop tests. Table 2.6.7.4.2-2 of the application summarizes the LS-DYNA calculated impact limiter crush depths and cask rigid-body decelerations for the hot and cold temperature conditions. Since all calculated decelerations for the cold, governing, condition bound the corresponding test equivalent values obtained from the 1/4-scale model drop tests, as can be seen in the table below, the LS-DYNA results are conservative.

Calculated and Test Equivalent Peak Decelerations - Prototype, 30-Ft Drop

	Rigid-Body Deceleration (g)			
	Calculated		**1/4-Scale Model**	
Drop Orientation	Hot (200°F)	Cold (-40°F)	Test Equivalent	Design
Side	42.9	48.5	41 (=164 x 1/4)	55.0
Top/Bottom End	40.8	39.9	30.5 (=122 x 1/4)	48.0
C.G. Over Corner	24.2	36.0	31.5 (=126 x 1/4)	48.0

On the basis of the above, the staff agrees that the 20-g design basis deceleration selected by the applicant bounds the 1-ft free-drop tests for performing structural evaluations of the package. Similarly, design decelerations of 55 g, 48 g, and 48 g, as listed above, bound the 30-ft side, top/bottom end, and c.g.-over-corner free drop tests, respectively.

2.3 Structural Performance

The applicant continued to use the analysis approaches, such as the ANSYS linear elastic analysis, and acceptance criteria for the approved NAC-STC to perform the structural evaluation of the amended system. In the following, the staff reviews the structural performance of the design features unique to the amended system, considering the free drop design decelerations and load combinations with other effects, such as bolt preload, thermal expansion, and internal pressures.

2.3.1 Lifting and Tiedown Standards

Lifting Devices. For an increased package weight to 260,000 lbs, the applicant reanalyzed the lifting trunnions and demonstrated that stresses in the trunnion shank,

trunnion base weld, and weld/forging interfaces are less than 1/6 and 1/10 of the material yield and ultimate strengths, respectively. This satisfies the NUREG-0612 non-redundant lifting criteria, which exceed those of 10 CFR 71.45(a).

Six 2-inch diameter hoist rings are used with a pair of three-legged slings in the redundant lifting of the canister. At a canister weight of 66,000 lbs and minimum sling angle of 60° from the horizontal, the maximum load in the hoist ring is less than the rated capacity of 30,000 lbs.

Tiedown Devices. The applicant increased the stresses in the previously approved NAC-STC by 4% to obtain those in the amended system weighing up to 260,000 lbs. The resulting stress margins are positive, which satisfy the requirements of 10 CFR 71.45(b).

2.3.2 Normal Conditions of Transport

Cask Body. For an increased package weight to 260,000 lbs and for load combination effects, such as closure lid bolt preload, cask internal pressure, and thermal expansion, the applicant followed the same approaches used for the previously approved NAC-STC to perform structural analysis of the cask.

The applicant determined that stresses in the cask body are within the allowable for the hot and cold conditions, consisting of ambient temperatures of 100°F and -40°F, respectively. Sections 2.6.7-1, -2, and -3 of the application present analyses for the end-, side-, and corner-drop conditions, respectively, with acceptable results.

Tables 2.6.7.5-1 through -4 of the application reevaluate the inner and outer lid bolts for a number of cask drop orientations under the thermal hot and cold conditions, for the amended cask content weight of 77,885 lbs. At a maximum cask internal pressure of 38.1 psig and a maximum bolt torque of 2,740 ft-lb, the inner lid bolt has a minimum stress margin of 0.06, which is greater than zero and acceptable.

Balsa Impact Limiter. For the penetration test provisions of 10 CFR 71.71(c)(10), the applicant calculated a depression of 0.57 inches for the impact limiter casing. This will not adversely affect performance of the package.

Transportable Storage Canister. Section 2.6.15 of the application presents the structural analysis of the canister with an increased overall length to 151.75 inches and increased bottom closure plate thickness to 1.75 inches. The analyses for the previously approved canister were repeated for the amended canister, but with a design basis deceleration of 20 g and applicable internal pressure and thermal loadings.

The applicant calculated stress intensities and stress margins at critical canister locations for individual and combined loading conditions. For the most critically stressed locations, the minimum safety margins for the primary membrane (P_m) and primary membrane-plus-bending ($P_m + P_b$) stresses are 0.03 and 0.26, respectively. The applicant also evaluated other structural failure modes, such as the bearing stresses of the canister resting on the cask inner shell, and the buckling capability the canister shell subject to an end impact, with acceptable safety margins.

Canister Spacer. The 12.7-inch high canister spacer for the amended NAC-STC is a weldment made of 3/8-inch stainless steel plates. It has a 70.6 inch-diameter circular base with six concentric cylindrical shells welded to it. Section 2.6.18.2 of the application demonstrates acceptable structural performance of the spacer for maintaining the canister in its analyzed configuration inside the cavity of the cask subject to an end-drop design deceleration of 20 g.

Fuel Basket. Section 2.6.16 of the application analyzes structural performance of the support disk, top and bottom weldments, and other structural components such as basket tie rods. For the support disk, four basket side drop orientations, at 0°, 38°, 75°, and 90° with respect to the cask axis, were considered. Similar to the evaluations performed for the previously approved system, stress results from the end and side-drop analyses were combined to obtain those for oblique drops at varied cask drop angles. The analysis demonstrated that the stresses in the fuel basket assembly and the buckling capabilities of the support disk are acceptable.

CY-RFA and CY-DFC. Section 2.6.17 of the application evaluates structural performance of the CY-RFA and CY-DFC for thermal effects and the free drop design decelerations of 20 g. The applicant continued to use the NUREG/CR-6322 criteria to demonstrate buckling strengths of the corner angles and fuel tube of the CY-RFA. The buckling strength of the CY-DFC fuel tube was evaluated with the Euler formula. The applicant also performed a finite element analysis of the CY-RFA grid plate. The structural performances of the CY-RFA and CY-DFC are acceptable.

CY-GTCC Waste Basket. Section 2.6.19.2 of the application evaluates the basket assembly against the requirements of ASME Code, Section III, Subsection NF. The applicant used the finite element method to perform stress analysis of the tube array and the shield shell weldments. For the tube array side drop analysis, a dynamic load factor (DLF) of 1.15 was applied on the cask rigid-body deceleration of 16.5 g to obtain an applicable static equivalent load of 19 g (16.5 x 1.15 = 19), which is bounded by the design deceleration of 20 g. The applicant considered four side drop orientations, at 45°, 60°, 75°, and 90°, with respect to the cask axis and determined that the 75° drop orientation governs the stress results. The minimum calculated stress margins in the tube array weldment are 0.18 for the primary membrane-plus-bending ($P_m + P_b$) stress, which occurs in the tube wall, and 0.22 for the weld primary membrane stress (P_m).

2.3.3 Hypothetical Accident Conditions

Cask Body. The applicant continued to consider an internal pressure of 50 psig, which bound the maximum of 38.1 psig for the amended cask, to perform the stress analysis of the cask. Other applicable loadings included an end drop deceleration of 56.1 g, which bounds the design deceleration of 48 g, and a side drop design deceleration of 55 g.

Sections 2.7.7.1, .2, .3, and .4 of the application present analyses for the end-, side-, corner-, and oblique-drop conditions, respectively, with acceptable stress results.

Tables 2.7.1.6-2 and -3 of the application reevaluate the inner closure lid bolt under the hot and cold conditions, respectively. At an internal pressure of 38.1 psig, a maximum bolt torque of 2,740 ft-lb, and an end drop design deceleration of 48 g, the lid bolt has a stress margin of 0.55. However, as demonstrated in Section 2.10.8.2.1, a minimum torque of 2,340 ft-lb must be applied to the bolt in order to prevent complete relief of bolt preload for the same end drop design deceleration.

Balsa Impact Limiter. By virtue of the model similitude laws, the 30-ft drop tests of the 1/4-scale model demonstrated that the balsa impact limiter of the amended NAC-STC will function satisfactorily to absorb drop energy and remain attached to the cask.

Transportable Storage Canister. Section 2.7.12 of the application presents structural analysis of the canister, including stress intensity and buckling capability results. Considering an end drop deceleration of 56.1 g, which bounds the design deceleration of 48 g, and the side drop design deceleration of 55 g, the applicant demonstrated structural adequacy of the canister. The corner- and oblique-drop results are bounded by the end- and side-drop results.

Fuel Basket. Section 2.7.13 of the application analyzes structural performance of the support disk, top and bottom weldments, and other structural components, such as the basket tie rod and split spacer. Deceleration loads at 56.1 g and 55 g were considered for the end- and side-drop tests, respectively. Similar to the evaluation of the previously approved system under normal conditions of transport, stress components from the end- and side-drop analyses were combined to obtain stresses for oblique drops at varied cask drop angles. The analysis demonstrated acceptable structural performance, including stress margins and buckling capabilities of the support disk.

CY-RFA and CY-DFC. Section 2.7.14 of the application considers a 60-g deceleration, which bounds the design decelerations, to perform end- and side-drop analyses of the CY-RFA and CY-DFC. The stress margins and buckling capabilities of the CY-RFA and CY-DFC are acceptable.

CY-GTCC Waste Basket. Section 2.7.10.1 of the application evaluates the basket assembly against the requirements of ASME Code, Section III, Subsection NF. Only the shield shell weldment is required to maintain structural integrity during accident conditions of transport. For the side drop, due to a lack of circular symmetry of the load paths between the shield shell rib supports and the tube array corners which rest on the shell wall, the applicant considered four shield shell drop orientations (45°, 60°, 75°, and 90°) with respect to the basket axis. At the side-drop design deceleration of 55 g, the minimum stress margins are 2.74 for the primary membrane-plus-bending ($P_m + P_b$) stress and 0.14 for the weld primary membrane stress (P_m).

Fuel Tube. Section 2.7.13.4 of the application evaluates structural performance of the fuel tubes for the end drop deceleration of 55 g, which bounds the cask design deceleration of 48 g, and the side drop of 60 g, which bounds the design deceleration of 55 g. For the end impact, the applicant calculated a bearing stress margin of 3.46. For the side impact, the applicant performed non-linear finite element analyses of the fuel tube for two loading cases: the pressure loading to simulate a uniform distribution of the fuel assembly weight and the grid loading a line load exerted through the grid spacer.

The total strains calculated are 0.019 for the pressure loading and 0.047 for the grid loading. These values correspond to the strain margins of 9.50 and 3.26, respectively. Considering the acceptable strain to be one half of the material failure strain of 0.4 in/in, these strains are acceptable.

Fuel Rod Buckling. Section 2.9.3 of the application evaluates fuel rod buckling for four fuel assembly types: Westinghouse Zirc-4, Westinghouse SS304, B&W Zirc-4, and B&W SS304. Similar to the fuel rod evaluation approach for the previously approved NAC-STC, the amended application includes pellet weights and derated material properties of irradiated fuels for calculating clad section properties. By considering a cask end impact of 55 g, which is impulsive and short duration and bounds the design deceleration, the applicant calculated dynamic load factors for the four rod types. This results in static equivalent g-loads all less than the corresponding first-mode buckling capabilities of the fuel rods, demonstrating that the fuel rods will not buckle for a 30-ft end drop accident.

Based upon the review of the information and analyses presented in the application, the staff concludes that the amended NAC-STC system is structurally capable of meeting the 10 CFR Part 71 package performance requirements.

3.0 THERMAL EVALUATION

The objective of this review is to verify that the thermal performance of the NAC-STC-Connecticut Yankee Amendment transportation package has been adequately evaluated for the tests specified under normal conditions of transport and hypothetical accident conditions and that the transportation package design (canister, cask, impact limiters, and personnel barrier) satisfies the thermal requirements of 10 CFR Part 71.

3.1 Description of the Thermal Design

3.1.1 Design Features

The thermal design features of the package include the following:

a. The Multi Purpose Canister (MPC) is designed to store up to 26 Connecticut Yankee fuel assemblies, reconfigured fuel assemblies and damaged fuel in damaged fuel cans, and is referred to as CY-MPC. This right-circular stainless-steel cylinder has overall dimensions of about 69-inches internal diameter, 71-inches external diameter, and 152-inches height. Out of the 26 loading positions, 4 present slightly oversized dimensions.

b. Greater Than Class C (GTCC) waste may also be stored in a canistered mode. The CY-MPC canister with the GTCC basket accommodates up to 24 fuel assembly sized containers of GTCC waste. These canisters have the same external dimensions as the respective spent fuel canisters, but have different basket designs.

c. The fuel basket design is a right-circular cylinder with square stainless-steel fuel tubes laterally supported by 28 stainless steel equally spaced support disks (0.5-inch thick and 69.15-inch in diameter). Inter spaced among the support disks are 27 aluminum-alloy heat transfer disks (0.5-inch thick and 68.87-inch in diameter).

c. Helium gas is used to backfill the canister to an internal pressure of 1 atm (normal operation) in order to enhance heat transfer within the canister.

e. The Storage Transport Cask (STC) cavity space is 71-inches in diameter and 165-inches in height. The CY-MPC canister is located in the STC cavity by a single stainless steel spacer formed by a series of concentric rings welded to a base plate. The spacer is installed below the canister in order to provide support.

f. The 0.18-inch annulus between the loaded canister and the transportation cask is backfilled with helium gas at 1 atm to enhance heat transfer from the canister.

g. The side wall region of the transportation cask consists of a series of concentric cylinders or shells. The 1.5-inch stainless steel inner shell is surrounded by a 3.7-inch thick cylinder of chemical copper lead for attenuating gamma radiation. Another 2.65-inch stainless steel outer shell surrounds the lead gamma shield and provides additional structural support. Neutron shielding is provided by a 5.5-inch thick shell of NS-4-FR material surrounding the outer shell. The most external layer is a 0.25-inch thick stainless steel neutron shield shell enveloping/protecting the resin-like neutron shield material.

h. Twenty four 0.55-inch (14 mm) thick copper/stainless steel heat transfer fins are equally distributed radially on the outer shell. The fins are 0.315-inch (8 mm) thick stainless steel plate explosively bonded to 0.236-inch (6 mm) thick copper plate. These fins are spaced lengthwise between the cask outer shell and the neutron shield shell to enhance heat transfer through the solid NS-4-FR neutron shield material to the ambient environment. The relatively high thermal conductivity of the neutron shield shell then distributes the heat around the exterior of the cask for efficient transfer to the ambient environment.

i. The outer surface of the neutron shield shell is exposed to the ambient conditions. No special coating is used.

j. The impact limiters consist of a combination of redwood and balsa wood encased with an outer skin of stainless steel. They are assumed to thermally insulate the cask ends from the ambient environment.

k. The personnel barrier is fabricated of an aluminum mesh, with approximately 12.8 inches of clearance between the cask exterior and the barrier. The design provides a minimum 60% free opening, which

yields a nearly unobstructed flow of air around the transportation cask for convective as well as radiative heat transfer to the ambient environment.

3.1.2 Contents Decay Heat

The following table indicates the envisioned heat load limitations of the NAC-STC cask when holding Connecticut Yankee fuels and/or GTCC waste:

	SNF[1]	**RFA**[2]	**DFC**[3]	**GTCC**[4]
Maximum Assembly Decay Heat (kW)	0.654	0.321	0.654	0.208
Maximum Cask Decay Heat (kW)	17.0	17.0	17.0	5.0

1 - Intact Spent Nuclear Fuel Assemblies

2 - Reconfigured Fuel Assemblies must be loaded in one of the 4 oversized fuel loading positions

3 - Damaged Fuel Cans must be loaded in one of the 4 oversized fuel loading positions

4 - GTCC assembly sized waste containers have a different basket design

Connecticut Yankee 15x15 PWR spent fuel includes Westinghouse, Babcock & Wilcox, Gulf General Atomic, and NUMEC designs. Some of the early Westinghouse designs used stainless steel, instead of zircaloy, for the cladding material. The Westinghouse Vantage 5H requires a special basket design, where the top weldment only allows 24 fuel tube penetrations.

For thermal analysis, a bounding axial power distribution is assumed for every loaded spent fuel assembly, as shown in Figure 3.4-27 in the SAR. A maximum peaking factor of 1.1 is considered.

3.1.3 Summary Tables of Temperatures

Tables 3.4-1 and 3.4-2 in the SAR show the maximum temperatures of the major NAC-STC components, the canister, the fuel basket, and fuel rods obtained when simulating Normal Conditions of Transport (NCT). Table 3.4-4 in the SAR shows the comparison between maximum allowable component temperatures and the peak values obtained from NCT simulations. Extra cask component maximum temperatures under NCT are provided in Table 3.4-5 in the SAR. For the Hypothetical Accident Conditions (HAC), Table 3.5-1 in the SAR shows the maximum temperature values and the time at which they occurred during the 30 minutes fire and post-fire simulations. Figure 3.5-3 in the SAR shows the time evolution for the main NAC-STC components during the fire

transient. These temperature values are consistent with those presented throughout the SAR for both NCT and HAC. All components remained below their allowable limits under NCT and HAC.

3.1.4 Summary Tables of Maximum Pressures in the Containment System

The pressure calculations of the containment system (canister and transportation cask) under NCT and HAC were reviewed and found consistent with the pressures presented throughout the appropriate sections of the SAR.

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

The application provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the cask. Also provided are the thermal emissivities used to model the radiative heat transfer to and away from appropriate internal and external surfaces within the package. The thermal properties used for the analysis are taken from sound technical references and are appropriate for the materials specified. Additionally, the fluid properties of the surrounding air and the backfill helium were provided for the evaluation of thermal conduction and convection parameters. All the provided properties are appropriate for the conditions of the cask required by 10 CFR Part 71.

3.2.2 Technical Specifications of Components

References for the technical specifications of pre-fabricated package components for O-rings and neutron absorber materials were provided by the applicant. All components were shown to satisfactorily perform under NCT and HAC.

3.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of cask containment, radiation shielding, and criticality were specified.

3.3 Thermal Evaluation under Normal Conditions of Transport

3.3.1 Model

The thermal analysis for the CY-MPC package uses three ANSYS finite element models:

a. A three-dimensional model is used to thermally evaluate the cask in a horizontal position with the fuel basket in contact with the canister, which is, in turn, in contact with the cask inner shell. The model's comprised of the fuel assemblies, fuel tubes, support and heat transfer disks, canister shell, lids, and bottom plate, spacer at the bottom of the canister, and NAC-STC. The NAC-STC portion of the model includes

the inner and outer shell, lead gamma shield, inner and outer bottom plates, inner and outer lids, upper and lower neutron shields, radial neutron shield, and neutron shield shell. The fuel regions and the fuel tubes with BORAL plates in the three-dimensional model are represented using effective conductivities.

b. The effective conductivity of the fuel is determined by a second ANSYS model, which is a detailed two-dimensional thermal model of the fuel assembly. The model includes the fuel pellets, cladding, and helium gas occupying the gap between fuel pins and the gap between fuel pellets and cladding.

c. A third ANSYS one-dimensional model is used to calculate the effective conductivity of the tube wall, BORAL plate, and cladding.

The following conservative assumptions are noted:

a. The top and bottom ends of the STC model are considered to be adiabatic since no heat transfer occurs between the ends of the cask and the environment due to the presence of the impact limiters. The impact limiters are not represented in the thermal model.

b. Small gap may exist between the lead shielding and the outer shell, due to differential thermal expansion of the two concentric layers. Air is assumed to be the filler for this gap.

3.3.2 Limiting Conditions

The following steady-state conditions were simulated with the ANSYS computer code:

Case 1: T_{air} = 100°F, with insolation, maximum decay heat

Case 2: T_{air} = -40°F, no insolation, maximum decay heat

The MPC was assumed to be loaded with 26 fuel assemblies, each generating 0.654 kW (guaranteeing the maximum heat load of 17 kW per cask), with the axial power distribution shown in Figure 3.4-27 in the SAR. The results obtained from this bounding approach cover the envisioned uses of the MPC for holding intact fuel as well as damaged and reconfigured fuel assemblies, due to their limited heat load and special location within the basket. A thermal analysis for the GTCC basked was also performed.

Tables 3.4-1 and 3.4-2 in the SAR summarize the results from the limiting condition calculations (Cases 1 & 2, respectively). Table 3.4-4 in the SAR shows the comparison between maximum allowable component temperatures and the peak values obtained from the NCT simulations. The most limiting component is the neutron shielding material (NS-4-FR) with a safety margin of 12°F. The safety margin for the cladding material was 19°F. Under NCT, all of the materials used remained below their respective allowable temperatures.

The applicant provided a sensitivity study where the effect of variations in emissivity, convection heat coefficient, and dimensions due to manufacturing tolerances were analyzed. The results indicate that the maximum increase in fuel cladding temperature is 5°F. All component temperatures remained within their respective allowable temperature ranges.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(1) and (2).

3.3.3 Accessible Surface Temperature

Under NCT, the package is enclosed by a protective screen to ensure that the accessible surface remains well below a temperature of 185°F. The ANSYS model suggested by the applicant indicates the personal barrier temperature to be near 140°F or below.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.43(g) and must be shipped as exclusive use.

3.3.4 Maximum Normal Operating Pressure (MNOP)

The applicant uses a bounding assembly design with a burnup of 43,000 MWd/MTU and backfill pressure of 475 psig. The MNOP is based on the initial cask Helium backfill, the canister backfill, fuel rod fill gas, fission gases, PWR control component gases, and no containment by the canister shell pressure boundary. Conservatively, 100% of the fuel rods are considered failed. For each postulated rod failure, 100% of the rod fill gas, 30% of the fission gases, and 30% of the gases generated within the PWR control rods are assumed to be released into the cavity. The helium temperature is conservatively assumed to be 450°F, instead of the 402°F obtained from the NCT thermal analyses. The MNOP in the canister was determined to be 42 psig. Assuming an absent canister, the MNOP was determined to be 38 psig.

3.3.5 Confirmatory Analyses

The confirmatory analyses performed by the staff involved verifying the effective conductivity values for fuel assembly, fuel tube, and neutron shield by either developing ANSYS models or comparing against other applications and publications. The staff found the fuel assembly conductivity values to be fairly close to the Wooton-Epstein predictions (which assume 17x17 assembly in air), but almost a factor of 2 smaller than the values described in the TRW study[1] for DOE. This degree of conservatism guarantees a well bounded calculation for the fuel cladding temperatures.

3.3.6 Summary

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71.

3.4 Thermal Evaluation under Hypothetical Accident Conditions

3.4.1 Model

The same ANSYS thermal three-dimensional model used for steady-state evaluations was applied, including the presence of the impact limiters and the adiabatic treatment of the cask ends. The following additional conservative assumptions are noted:

a. Credit is taken for the presence of NS-4-FR in the neutron shield during the fire, but the resin material is replaced by “air” during the cool down period. This results in a decrease in equivalent conductivity, dampening the heat removal during the post-fire period.

b. The neutron shield shell external surface emissivity and absorptivity are assumed to be 0.9 during the fire and 0.36 in the post-fire period.

c. Insolation does not occur during the 30-minutes fire, but is accounted for during the post-fire period.

d. The personnel barrier is assumed to be lost.

Following the 30-minute fire event, the transient analyses are continued for a sufficient time (29.5 hours) to determine the maximum temperatures reached for all components.

3.4.2 30-Minute Thermal Test

Table 3.5-1 in the SAR summarizes the results from the fire simulation. Peak temperature values and the time of their occurrence are provided for the transportation cask main components. Also provided are the allowable material limits. Figure 3.5-3 in the SAR shows the time evolution for some of the component temperatures during the transient.

Substantial thermal margins are observed during the fire transient, with peak temperatures even below the allowable short-term values. These findings were expected, due to the large thermal masses displayed by the transportation cask, and the NS-4-FR neutron shield layer which acts like a radiation shield to limit the amount of heat transferred into the cask during the 30-minutes event. The canister wall sees a temperature rise of less than 40°F.

3.4.3 Maximum Internal Pressure

Using the same assumptions from the MNOP calculation except for the Helium temperature, which is conservatively assumed to be 750°F instead of the calculated 462°F, the applicant finds the maximum canister internal pressure to be 61 psig. Assuming an absent canister, the applicant calculated a maximum internal pressure to be 56 psig.

3.4.5 Confirmatory Analyses

The confirmatory analyses involved verifying the effective conductivity values for fuel assembly, fuel tube, and neutron shield by either developing ANSYS models or comparing against other applications and publications. The staff found the fuel assembly conductivity values to be fairly close to the Wooton-Epstein predictions (which

assume 17x17 assembly in air), but almost a factor of two smaller than the values described in the TRW study[1] for DOE. This indicates a conservative approach used by the applicant.

3.4.6 Summary

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(4).

3.5 Evaluation Findings

The staff agrees with the applicant's approaches to modeling the transportation package and the results that were obtained.

Based on review of the statements and representations in the application, the staff concludes that the design has been adequately described and evaluated and that the package meets the thermal performance requirements of 10 CFR Part 71.

3.6 References

1. "Spent Nuclear Fuel Effective Thermal Conductivity Report," Document Identifier BBA000000-01717-5705-00010 REV 00, Prepared by TRW Environmental Safety Systems, Inc., for the U.S. Department of Energy, July 11, 1996

4.0 CONTAINMENT EVALUATION

The NAC-STC contents have been amended to include Connecticut Yankee (CY) PWR intact and damaged fuel and canistered fuel debris within a Transportable Storage Canister (TSC). NAC also modified the definitions of the primary containment boundaries which are provided below.

Containment Condition A: The containment boundary for the transport of directly loaded intact PWR spent fuel assemblies following extended storage of the cask at an ISFSI licensed in accordance with 10 CFR Part 72. The primary containment boundary consist of: inner shell; upper and lower shell rings; bottom inner forging; top forging; inner lid; inner lid outer metal o-ring; inner lid interseal test port threaded plug with metal o-ring; vent port cover plate; vent port outer metal o-ring; vent port interseal port threaded plug with metal o-ring; drain port coverplate; drain port coverplate outer metal o-ring; and drain port coverplate interseal test port plug with metal o-ring.

Containment Condition B: The containment boundary for the transport of (1) directly loaded intact PWR spent fuel assemblies loaded immediately prior to transport; or (2) canistered Yankee Class or Connecticut Yankee spent fuel assemblies, Reconfigured Fuel Assemblies, Damaged

Fuel Cans or GTCC waste loaded into the NAC-STC immediately prior to transport.

The primary containment boundary consists of: inner shell; upper and lower shell rings; bottom inner forging; top forging; inner lid; inner lid inner metal o-ring; vent port cover plate; vent port inner metal o-ring; drain port coverplate; and drain port coverplate inner metal o-ring.

The TSC is designated as the separate inner container (i.e., a secondary containment system) to meet the requirements of 10 CFR 71.63 when shipping damaged fuel or fuel debris. The secondary containment boundary includes the TSC shell, bottom baseplate, shield lid, vent and drain port covers and their associated welds, and the structural lid.

The staff evaluated the modified definitions for their impact on the overall containment requirements.

The staff determined that the 10 CFR 71.63(b) criteria for a separate inner container are met. Since the TSC is completely welded closed and leaktight in accordance with the requirements of ANSI N14.5-1997, it will adequately contain the Connecticut Yankee damaged fuel and fuel debris for the following reasons: 1) During normal and hypothetical accident conditions, there is reasonable assurance that the TSC maintains its structural integrity as indicated in the SAR Chapter 2; 2) the peak containment boundary component temperatures and pressures are within the design-basis limits, as indicated in SAR Chapter 3; and 3) the integrity of the TSC closure welds are assured through nondestructive examinations.

The staff concludes that the definitions for the primary and secondary containment boundaries will not compromise the safety of the cask. Also, the TSC is an acceptable separate inner container for transportation of Connecticut Yankee damaged fuel and fuel debris, and meets the requirements of 10 CFR 71.63 (b).

The leak test acceptance criteria has not been revised. The primary containment boundary for directly loaded fuel, and the primary and secondary containment boundaries for the shipment of the TSCs will be tested to meet the leak tight criteria of ANSI N14.5-1997, i.e., a maximum allowable leakage rate of 1×10^{-7} ref-cm^3/sec.

The following leakage rates and test sensitivities, as indicated in the SAR Table 4.1-1, will be used for each containment boundary:

	Primary Containment Boundary (NAC-STC Transport Cask)	Secondary Containment Boundary[1] (Transportable Storage Canister)
Containment Boundary A		
"As tested" leakage	$\leq$ 2 x 10^{-7} cm^3/sec (helium)	not required
Test sensitivity	$\leq$ 1 x 10^{-7} cm^3/sec (helium)	not required
Containment Boundary B		
"As tested" leakage	$\leq$ 2 x 10^{-7} cm^3/sec (helium)	$\leq$ 2 x 10^{-7} cm^3/sec (helium)
Test sensitivity	$\leq$ 1 x 10^{-7} cm^3/sec (helium)	$\leq$ 4 x 10^{-8} cm^3/sec (helium)

[1] Secondary Containment Boundary only required when shipping damaged fuel or fuel debris.

The staff evaluated whether the leak test acceptance criteria would meet 10 CFR 71.51(a) and 10 CFR 71.51(a)(2) for the Connecticut Yankee contents. Since the NAC-STC containment boundaries will be tested to the leaktight standard of ANSI N14.5-1997, no credible release of radioactive material is expected under any transport condition. Therefore, no calculation or determination of the releasable radiological source term and the corresponding dose consequence were necessary to demonstrate compliance with 10 CFR Part 71 requirements. Results of the structural and thermal analyses (SAR Chapters 2 and 3, respectively) showed that, under normal conditions of transport, the containment system remains leaktight when subjected to conditions and tests specified in 10 CFR 71.71. Therefore, loss or dispersal of radioactive material from the cask will be less than the limits of 10 CFR 71.51(a)(1). Similarly, the containment system is expected to remain intact if subjected to the hypothetical accident conditions of 10 CFR 71.73. Because there are no credible events that will compromise the integrity of the containment boundaries under hypothetical accident conditions, the limits of 71.51(a)(2) are also met.

In addition to revising the containment boundary definitions, NAC revised the leak testing methods which are described in Chapter 7 of the SAR. The staff's evaluation of the revised leak testing procedures is discussed under "Operating Procedures."

Based on review of the statements and representations in the application, the staff concludes that the design has been adequately described and evaluated and that the package meets the containment performance requirements of 10 CFR Part 71.

5.0 SHIELDING EVALUATION

5.1 Cask Contents

The NAC-STC transportation cask is designed to hold up to 26 CY fuel assemblies, reconfigured fuel assemblies and damaged fuel contained in CY-MPC damaged fuel cans. The STC will also hold up to 24 fuel assembly-sized canisters containing greater than class C (GTCC) waste. A bottom spacer is used when the canister containing CY fuel or GTCC waste is placed in the STC cavity.

CY spent fuel consists of 15x15 PWR stainless steel and Zircaloy-clad fuel assemblies with a square cross-section. There are two basket configurations that will be used for the CY fuel in the STC. One configuration is the 26-assembly basket and the second is the 24-assembly basket.

Reconfigured fuel assemblies will also be loaded into the STC. The reconfigured fuel assembly is a stainless steel 10x10 array of tubes attached to upper and lower end fittings that are similar to those used on standard assemblies. Damaged or bowed fuel rods are inserted into the tubes. The design allows the release of gaseous products and liquids but minimizes the dispersal of particulates.

CY-MPC damaged fuel can will hold a complete fuel assembly which may be intact or damaged. The damaged fuel can is fabricated from stainless steel and has both top and bottom closures that allow the release of gaseous products and liquids but minimizes the dispersal of particulates.

Some of the CY assemblies will be stored with flow mixers or with reactor control cluster assemblies installed. Flow mixers are thimble plug assemblies used during reactor operation to maintain equal coolant flow in fuel assemblies that do not contain a reactor control cluster. Reactor control clusters are used to control the reactivity of the reactor during operations and shutdown.

GTCC consists of sections of the core baffle, core barrel, core support plate, and miscellaneous related hardware associated with these components. The major components were hydro-laser cut underwater into pieces of a size that are loaded into containers that have the same external dimensions as a CY fuel assembly. Dross material will be disposed of as low-level waste. Principle isotopic constituents of the GTCC waste are presented in SAR table 1.2-6.

The design of the STC has not changed and therefore will not be discussed.

5.2 Source Term

CY fuel assemblies were manufactured by Westinghouse, Gulf Nuclear/Gulf General Atomic, NUMEC, and Babcock & Wilcox. The 26-assembly basket is used for all of the fuel types except Westinghouse Vantage 5H fuel. The Vantage 5H fuel must be placed in the 24-assembly basket. The 24-assembly basket can also be used for the other CY fuel. The 24-assembly basket is used for loading zircaloy clad fuel with an enrichment

above 3.93 wt. percent ^{235}U or stainless steel clad fuel with an enrichment above 4.03 wt. percent ^{235}U.

The design basis fuel is the Westinghouse manufactured fuel with a maximum burnup of 38,000 MWd/MTU, minimum initial enrichment of 3.65 wt. percent ^{235}U and 10 year cooling time. Some of the stainless steel clad fuel has a minimum initial enrichment as low as 3.0 wt. percent ^{235}U, however the maximum burnup of these assemblies is around 30,000 MWd/MTU.

The gamma source term is made up from the fission products in the fuel and activation products in the fuel assembly hardware. The source term from the fuel assembly hardware is primarily due to neutron activation of ^{59}Co impurities in the stainless steel. ^{59}Ni and ^{58}Fe provide minor contributions to the gamma source term of the hardware. NAC used the value of 0.5 g/kg for the amount of ^{59}Co impurity for stainless steel clad fuel and the value 1.2 g/kg for zircaloy clad fuel. The value of 1.2 g/kg ^{59}Co impurity level is the generally accepted value for calculating the source of the hardware. However, NAC submitted information indicating the ^{59}Co impurity level for stainless steel clad fuel did not exceed 0.5 g/kg. Therefore, the use of 0.5 g/kg was determined to be acceptable for the CY stainless steel clad fuel. Based on this information, the in-core flux spectrum was determined using the SAS2H computer code.

The neutron source is from actinide spontaneous fission and from alpha-neutron reactions with the oxygen in the UO_2 fuel. The isotopes ^{242}Cm and ^{244}Cm are the major contributors to the source term. The neutron spectra was determined using by the SAS2H and ORIGEN-S computer codes.

The CY GTCC source term was determined by characterization of the waste to identify the maximum source component in curies/gram. The core baffle was determined to be the maximum source component. The code ORIGEN-S was used to calculate the core baffle radionuclide inventory, decayed over a period of 10-years.

5.3 Dose Rates

MCBEND, which is a three-dimensional Monte Carlo code, was used to determine the dose rates for the STC. MCBEND uses a fractal geometry system which allows detailed modeling of the source term, basket and cask, including streaming paths. The STC was modeled under both NCT and HAC.

Hypothetical accident conditions modeled include: top axial, bottom axial, and radial lead slump, loss of radial neutron shield and shield shell, and loss of upper and lower impact limiters.

The following table presents the dose rates for the CY fuel under NCT and HAC.

Connecticut Yankee Fuel Maximum Dose Rates under Normal Conditions of Transport (mrem/hr)					
		Stainless Steel Clad Fuel		Zircaloy Clad Fuel	
Location	Source	Surface	2-meter	Surface	2-meters
Top Axial	Neutron	0.1	0.1	0.2	0.1
	Gamma	0.1	0.1	0.2	0.1
	Total	0.3	0.1	0.4	0.2
Radial	Neutron	30.2	0.9	43.6	1.3
	Gamma	3.8	2.7	5.5	2.3
	Total	34.0	3.6	49.1	3.6
Bottom Axial	Neutron	0.5	0.1	0.8	0.1
	Gamma	1.2	0.2	1.2	0.2
	Total	1.6	0.3	2.0	0.3
10 CFR 71 Limits	Total	200.0	10.0	200.0	10.0

Connecticut Yankee Fuel Maximum Dose Rates Under Hypothetical Accident Condition (mrem/hr)					
		Stainless Steel Clad Fuel		Zircaloy Clad Fuel	
Location	Source	Surface	1-meter	Surface	1-meters
Top Axial	Neutron	2.1	9.5	3.1	14.1
	Gamma	0.2	1.2	0.3	1.3
	Total	2.4	10.8	3.4	15.4
Radial	Neutron	746.0	234.0	1123.0	348.0
	Gamma	60.0	25.0	47.0	21.0
	Total	806.0	259.0	1170.0	369.0
Bottom Axial	Neutron	4.7	12.3	7.6	17.6
	Gamma	3.8	0.8	3.7	0.9
	Total	8.5	13.1	11.3	18.5
10 CFR 71 Dose Rate Limit at 1 meter			1000.0		1000.0

5.4 Confirmatory Calculations

The staff performed confirmatory analyses of the gamma and neutron source terms for the design basis stainless steel and zircaloy-clad fuels. Staff used SAS2H and ORIGEN-S of the SCALE-4.4 computer code. Staff also reviewed the fuel parameters, material densities, and code input decks, and has reasonable assurance that the design basis gamma and neutron source terms are adequate for the shielding analysis.

5.5 Evaluation Findings

Staff has reviewed the information contained in the SAR and responses to RAIs, has performed confirmatory evaluations of the source terms for the design basis stainless steel and zircaloy clad fuel, and has reviewed the input data for the dose rate calculations.

Based upon the information provided by the licensee and the confirmatory calculations, staff has reasonable assurance that the dose rates determined by NAC are representative of dose rates which would occur when the transfer cask and the concrete cask are filled with design basis stainless steel or zircaloy fuel.

6.0 CRITICALITY EVALUATION

The staff reviewed the criticality analysis to ensure that the transportation of Connecticut Yankee spent fuel in the CY-MPC canister within the NAC-STC system meets the criticality safety requirements of 10 CFR Part 71.

6.1 Description of Criticality Design

The applicant revised the design of the STC system to include the CY-MPC canister, with up to 26 Connecticut Yankee 15 x 15 PWR fuel assemblies, as allowable contents. Criticality safety of the CY-MPC basket depends on the geometry of the fuel basket and the use of fixed Boral panels for absorbing neutrons. The basket features square fuel tubes, each with Boral panels fixed to the four outer walls. The primary design parameters that ensure subcriticality are the minimum flux trap width between fuel tubes and the minimum ^{10}B content of 0.02 g/cm^2 in each of the Boral panels. The flux trap widths in the CY-MPC basket vary from 1.00 to 3.50 inches. Section 6.3 provides sketches of the CY-MPC basket.

The basket is contained within a cylindrical stainless steel MPC canister, which is placed inside the STC transportation cask. The transportation cask consists of thick inner and outer steel shells, with lead gamma shielding in the annulus between them, and an outer NS-4-FR neutron shield contained in a thin steel shell. Prior to transport, the CY-MPC canister will be contained in the transfer cask, which is also considered in the criticality analysis. The transfer cask consists of concentric shells of steel, lead, NS-4-FR, and steel.

The staff has reviewed the description of the packaging design and concludes that it provides an adequate basis for the criticality evaluation.

6.2 Spent Nuclear Fuel Contents

The NAC-STC system with the CY basket is designed to transport up to: 1) 26 zircaloy clad spent fuel assemblies enriched up to 3.93 wt. percent ^{235}U, 2) 26 stainless steel clad spent fuel assemblies enriched up to 4.03 wt. percent ^{235}U, or 3) 24 zircaloy clad spent fuel assemblies enriched up to 4.61 wt. percent ^{235}U. The 24- and 26-assembly baskets are identical, except that 2 guide tubes of the 26-assembly basket are mechanically blocked to create the 24-assembly basket. In addition, up to four Connecticut Yankee reconfigured fuel assemblies (RFAs) or damaged fuel canisters (DFCs) are permitted, when placed in the four oversized corner guide tubes.

The applicant divided the Connecticut Yankee spent fuel inventory into four basic fuel groupings. Table 6.2-4 of the amendment lists the fuel assembly parameters for each of the four fuel groupings. The fuel assembly parameters that determine fuel grouping are:

- maximum initial enrichment (wt% ^{235}U)
- maximum MTU
- cladding material (Zircaloy or stainless steel)
- maximum fuel rod pitch
- maximum pellet diameter
- maximum active fuel length
- minium rod outer diameter
- minimum cladding thickness

Table 6.2-5 provides the RFA parameters. The RFA consists of a 10 x 10 array of stainless steel tubes, each containing one individual fuel rod, portions of an individual fuel rod, or loose fuel pellets and debris. The total equivalent fuel mass in each tube is restricted to that of one spent fuel rod. Other RFA parameters included in Table 6.2-5 are:

- maximum initial enrichment (wt% ^{235}U)
- maximum uranium mass (kg U)
- maximum tube pitch
- maximum tube outer diameter
- minimum tube thickness

The staff has reviewed the description of the spent nuclear fuel contents and concludes that it provides an adequate basis for the criticality evaluation.

6.3 General Considerations for Criticality

6.3.1 Model Configuration

The applicant explicitly modeled the CY-MPC basket inside the STC transfer cask, using the MONK8A code system[1]. The canister model includes the fuel assemblies, basket tubes, Boral sheets, structural and heat transfer discs, and water gaps between the discs. Fuel assembly hardware, such as spacers and top and bottom end fittings, are conservatively neglected. Sketches of the CY-MPC basket are provided in Section 6.3. The sketches are based on the engineering drawings provided in Section 1.3.2.

The calculation models conservatively assume the following:

1. fresh fuel isotopics,
2. 75% credit for the ^{10}B loading in the Boral panels, and
3. flooding of the fuel rod gap regions with pure water.

For analyses under both normal conditions of transport and hypothetical accident conditions, the applicant used reflective radial and periodic axial boundary conditions to represent an infinite, three-dimensional array of transfer casks. Analysis of an infinite array of STC transfer casks bounds the analysis of a single package required by §71.55. The cask model is the same for both normal conditions of transport and hypothetical accident conditions, since the cask and basket dimensions were not shown to be significantly different as a result of the tests specified in §71.73.

6.3.2 Material Properties

The compositions and densities for the materials used in the criticality safety analysis computer models are provided in Sections 6.3.3 through 6.3.6 of the CY-STC amendment. The minimum required areal density of the ^{10}B in the fixed neutron poison plates is 0.02 mg/cm^2. Section 8.1.7 of the SAR discusses the acceptance tests for Boral neutron absorber plates.

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.

6.3.3 Computer Codes and Cross Section Libraries

The applicant performed the CY-STC criticality analyses using the MONK8A criticality safety analysis code and the accompanying JEF 2.2-based point energy neutron library. The staff performed confirmatory analyses using the CSAS25 module of SCALE 4.4a (reference 2) with the 44-group cross-section library.

The MONK8A and SCALE 4.4a code systems are both widely used and acceptable for performing criticality analyses. The staff agrees that the codes and cross-section sets used are appropriate for this particular application and fuel system.

6.3.4 Demonstration of Maximum Reactivity

A number of parametric cases were analyzed to determine the most reactive model for spent fuel in the NAC-STC system. First, the applicant determined the most reactive assembly by varying assembly parameters within each fuel group. This analysis considered a full cask model containing 26 of each CY assembly group. The most reactive fuel assembly parameters were determined to be maximum fuel rod diameter, maximum fuel rod pitch, maximum active fuel length, minimum clad outer diameter, and minimum clad thickness. In addition, this analysis showed that 26 of the most reactive CY fuel assemblies loaded in the CY-STC produced a k-eff exceeding the calculated upper subcritical limit (USL). As a result, Zircaloy clad CY fuel assemblies exceeding enrichments of 3.93 wt. percent are limited to 24 per CY-STC.

The applicant then performed analyses to determine the most reactive mechanical configuration of the basket and contents. Fabrication tolerances found to have a positive effect on system reactivity were minimum neutron poison sheet width and

length, maximum fuel tube opening, and maximum spacer and heat transfer disc opening. The neutron poison sheet width was reduced beyond the fabrication tolerances to account for side-to-side shifting underneath the stainless steel cladding. An increase in system reactivity was also found to occur when the disc openings, guide tubes, and fuel assemblies were shifted together in specific radial directions. These directions are towards cask center for the 24-assembly configuration and towards guide tube positions 12 and 15 for the 26-assembly configuration (see Figures 6.4.4-1 and 6.4.4-2 of the SAR). Axial shifting of fuel assemblies due to a hypothetical accident conditions top-end drop are also considered. Structural analyses in Section 2 of the SAR show that the assembly top nozzles do not deform in such a scenario. However, the applicant assumes that the fuel within the fuel rods shifts into half the height of the rod plenum, exposing 1.905 inches of active fuel above the top of the neutron poison plates. Although no structural basis is provided for the fuel shifting into only half the plenum height, the applicant's analysis considers 4 inches of active fuel protruding beyond the neutron poison plates to provide margin on fuel shifting uncertainties. This resulted in a higher system reactivity, but maximum K-eff was still below the calculated USL.

The applicant also evaluated missing fuel rods from the CY fuel assemblies. This case used the most reactive 24-assembly basket configuration loaded with zircaloy clad fuel assemblies enriched to 4.61 wt. percent ^{235}U. The analysis removed up to 24 rods from each assembly, which optimized the H/U ratio and resulted in an increase in system reactivity. The applicant then evaluated this case limiting the assemblies with missing rods to the four corner locations. This calculation demonstrates that loading the assemblies with missing rods in corner locations does not significantly effect the overall reactivity of the CY-STC transfer cask system.

To bound the most reactive configuration of the damaged fuel canister, the applicant modeled the CY-STC 24-assembly basket loaded with 20 fuel assemblies enriched to 4.61wt. percent ^{235}U and with 4 damaged fuel assemblies in each corner location. Each damaged fuel can was modeled as a homogeneous mixture of fuel and water with varing volume fractions of each. An additional analysis was performed with the damaged fuel cans modeled as intact assemblies with 24 rods removed, and with the fuel/water mixture modeled either in between the remaining rods or outside the active fuel region. This analysis considered varying moderating conditions from full density to reduced density water, including uneven drain down conditions between the damaged fuel cans and the basket. The most reactive damaged fuel configuration was that with a fully flooded can and a loose fuel/water mixture located outside the active fuel region of an assembly with 24 missing rods.

The reconfigured fuel assembly was modeled in each guide tube of the CY-STC assuming several different configurations. The first configuration involved varying the pellet diameters within each RFA tube. The most reactive RFA was that with the maximum diameter of fuel in each tube. The second configuration varied the water density within the RFA. The fully flooded condition was found to be most reactive. The third configuration involved varying the water density of a homogeneous fuel/water mixture within each RFA tube. The most reactive condition was with the tubes full of fuel. The final configuration examined the effect of fuel debris in the RFA drain lines and drain reservoir in the bottom end fitting. This analysis showed that the most reactive condition was with solid fuel in the drain reservoir surrounded by water. None of these configurations were more reactive than the STC full of intact assemblies in either the 24- or 26-assembly configuration.

The results of the applicant's criticality analysis show that k-eff of the NAC-MPC system will remain below the calculated USL of 0.9425 for all allowed fuel loadings. The highest calculated k-eff was 0.9327 ± 0.0008 for the CY-STC transfer cask top-end drop under hypothetical accident conditions. The staff has reviewed the applicant's demonstration of maximum reactivity and agrees that the most reactive package condition has been identified.

6.3.5 Confirmatory Analyses

The staff performed confirmatory analyses using the Connecticut Yankee design basis fuel assembly parameters listed in Table 6.2-4, as well as cask design information provided in the engineering drawings in Section 1.3.2, and fabrication tolerance and mechanical perturbation information provided in Section 6.4.4.2.2. Using the CSAS25 sequence of SCALE 4.4a with the 44-group cross-section library, the staff modeled the CY-MPC canister in the STC transfer cask using assumptions similar to the applicant's. The staff's criticality analysis resulted in maximum system k-effs for the 24- and 26-assembly configurations that agreed with the applicant's analyses to within 1%, and were below 0.95 including calculation bias and uncertainty.

6.4 Benchmark Comparisons

The applicant ran MONK8A for selected critical experiments which were modeled by the code system developer. The critical experiments were selected based on their relevance to light water reactor fuel evaluations and to specific design components of the STC cask. Trending in k-eff was evaluated for enrichment, rod pitch, fuel pellet diameter, fuel rod diameter, H/U ratio, average group causing fission, ^{10}B loading in flux trap cases, and flux trap gap thickness. No statistically significant trends were discovered, with the largest correlation shown to be for k-eff versus flux trap gap thickness. The applicant used the method presented in NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," to calculate a USL of 0.9425 based on the data for k-eff versus flux trap gap thickness.

Although the applicant used an acceptable method to calculate the USL, it is not clear that the critical experiments were modeled using modeling techniques and code input options similar to those used in the STC analysis. Note that individual modeling techniques and selection of code input options are possible sources of uncertainty due to the analyst, and should be considered in the establishment of calculation bias and uncertainty. Therefore, the staff does not accept the benchmark evaluation provided by the applicant. The criticality analysis is accepted based on the facts that MONK8A is a well verified and widely accepted code for evaluating light-water-reactor fuel, and that extensive staff confirmatory analyses, performed using a different code and cross-section library, gave comparable results.

6.5 Findings

Based on the staff's review of the revised SAR and the staff's own confirmatory analyses, the staff concludes that the Model No. NAC-STC meets the criticality safety requirements of 10 CFR Part 71 for Connecticut Yankee fuel in the CY-MPC canister.

6.6 References

1. MONK8A, “A Monte Carlo Program for Nuclear Criticality Safety Analyses,” AEA Technology.

2. U.S. Nuclear Regulatory Commission, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” NUREG/CR-0200, March 1997.

7.0 OPERATING PROCEDURES

NAC provided updated Operating Procedures descriptions for the package. NAC also revised the leak test procedures and helium leak test acceptance criteria to appropriately reflect the testing of the Containment Condition A and B boundaries. The primary containment boundary for directly loaded fuel and the primary and secondary boundaries for the canistered transport of damaged spent fuel will be tested to meet the leak tight criteria of ANSI N14.5-1997.

The staff evaluated the contents of the generic procedures and guidance for operation of the NAC-STC. The staff concludes that the operating procedures and acceptance tests meet the requirements of 10 CFR Part 71 and that these procedures and acceptance tests are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

NAC revised the acceptance tests and maintenance program chapters to incorporate comments from utility member of the NAC Nuclear Technology Users Group, to clarify inspection and test requirements and to make the procedures consistent with other sections of the Safety Analysis Report. The majority of the Chapter 8 revisions are editorial. Significant changes to Chapter 8 are: 1) revision of the neutron shielding test to qualify installed NS-4-FR by chemical and physical testing of samples from the installed material batch, 2) performance of the thermal test of the NAC-STC cask body using a support structure for electric heaters rather than a fuel basket for each fabricated packaging and, 3) deletion of neutron absorption testing of the BORAL sheets.

The staff evaluated the acceptance tests and maintenance program for the NAC-STC considering the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices in its review. The staff concludes that the acceptance tests and maintenance procedures meet the requirements of 10 CFR Part 71 for the certification term.

CONDITIONS

In addition to the requirements of Subpart G of 10 CFR Part 71:

a. For Connecticut Yankee damaged fuel rods and fuel debris: if the total damaged fuel plutonium content of a package is greater than 20 curies, all damaged fuel shall be enclosed in a TSC which has been leak tested at the time of closure. The leak test shall have a test sensitivity of at least 1.0×10^{-7} cm^3/sec (helium) and shown to have a leak rate no greater than 2.0×10^{-7} cm^3/sec (helium).

CONCLUSION

The NRC staff concludes that the requested change will not affect the package's ability to meet the requirements of 10 CFR Part 71. Pursuant to 10 CFR Part 71, Certificate of Compliance No. 9235 for the NAC-STC transportation package is amended. All other conditions of Certificate of Compliance No. 9235 shall remain the same.

Issued on September 27, 2002.