CONF-890631--45

A Fuel Response Model for the Design of Spent Fuel Shipping Casks*

CONF-890631--45

DE89 013544

Anthony P. Malinauskas
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, Tennessee 37831-6135

Thomas A. Duffey
APTEK, Inc.
1257 Lake Plaza Drive
Colorado Springs, Colorado 80906

Robert E. Einziger
Battelle Pacific Northwest Laboratories
P.O. Box 999
Richland, Washington 99352

R. R. Hobbins
Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

Hans Jordan
Battelle Columbus Division
505 King Avenue
Columbus, Ohio 43201-2693

Y. R. Rashid and P. R. Barrett
ANATECH Research Corporation
San Diego, California 92038

Thomas L. Sanders
Sandia National Laboratories
Albuquerque, New Mexico 87185

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
Operated by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-84OR21400

*This work was performed at Sandia National Laboratories, Albuquerque, New Mexico, supported by the United States Department of Energy under Contract DE-AC04-76DP00789.

INTRODUCTION

The radiologic source terms pertinent to spent fuel shipping cask safety assessments are of three distinct origins. One of these concerns residual contamination within the cask due to handling operations and previous shipments. A second is associated with debris ("crud") that had been deposited on the fuel rods in the course of reactor operation, and a third involves the radioactive material contained within the rods. Although the lattermost source of radiotoxic material overwhelms the others in terms of inventory, its release into the shipping cask, and thence into the biosphere, requires the breach of an additional release barrier, viz., the fuel rod cladding. Hence, except for the special case involving the transport of fuel rods containing previously breached claddings, considerations of the source terms due to material contained in the fuel rods are complicated by the need to address the likelihood of fuel cladding failure during transport.

The purpose of this report is to describe a methodology for estimating the shipping cask source terms contribution due to radioactive material contained within the spent fuel rods. Thus, the probability of fuel cladding failure as well as radioactivity release is addressed.

APPROACH

The development of an appropriate source term methodology first requires the specification of the transport conditions (normal and accident) under which the methodology is to apply. These are defined in the United States Code of Federal Regulations, Title 10, Part 71 (10CFR71) [CFR 1987]. For spent fuel shipping casks, the most restrictive conditions for normal transport involve vibration, heat, and 1-ft. drops of the cask onto an unyielding, horizontal surface. Under accident conditions, the most significant events involve 30-ft. drops onto an unyielding, horizontal surface and the exposure to an engulfing, 800°C (1475°F) fire for a period of 30 minutes. The specification of these conditions then permits an evaluation to be made of the associated forces and thermal environments experienced by the fuel rods.

Determination of Transport Cask Environments

Thermal response analyses have been conducted [DUFFEY 1988] to estimate fuel rod cladding temperatures that might be experienced under the transport conditions stipulated in 10CFR71. These analyses indicate that, under normal conditions of transport, the peak fuel temperature of a PWR fuel assembly having a decay heat generation rate of 3 kw is predicted to be 315°C in a lead shielded truck cask. For a lead shielded rail cask transporting fuel assemblies of 1 kw decay heat generation per assembly, the peak fuel temperature is estimated to be 327°C under normal conditions of transport.

Transient analyses of the corresponding accident modes indicate a peak fuel temperature of 402°C for the truck case, and of 347°C for the rail cask.

Burian [BURIAN 1985] has indicated that only clad rupture by internal pressurization is a feasible mode of fuel cladding failure under the thermal environments stipulated by 10CFR71 and further, that a cladding temperature of about 725°C or above is required for this type of rupture to occur. In view of the significantly lower temperatures which are predicted to occur during shipment, failure of the fuel claddings due to elevated temperature is not considered further in this report.

Similarly, analyses have been made of effects due to shock and vibration normally experienced during both road and rail transport [DUFFEY 1988]; the resulting stresses likewise appear to be inconsequential relative to fuel cladding failure.

Thus, the problem of defining the shipping cask environments corresponding to the transport conditions stipulated in 10CFR71 reduce to considerations of the 1- and the 30-ft. cask drop events. Such drops can be described in a general way in terms of three phases: (1) the initial impact phase; (2) a pinned phase; and (3) a slap down phase. End-on (90° from horizontal) and side-on (0°) cask drops are characterized by only the initial impact phase, whereas all other orientations for which the projection of the center of mass of the cask lies outside the initial impact footprint will experience the pinned and slapdown phases as well. These are referred to as corner drops.

Analysis of the drop events was performed using a rigid body kinematics model, SLAM (Spent fueL cask impact Analysis Method), which was developed [DUFFEY 1988] expressly for the present study. In brief, the model treats each cask as a rigid body with deformable impact limiters. In the initial impact phase, crushup of the impacting end occurs and a small amount of rotation of the cask is permitted. This phase terminates when the velocity of the initial impact end becomes zero.

No crushing of the impact limiters occurs during the pinned phase, although slippage of the original impact surface may be experienced. This phase terminates when the upper impact limiter contacts the unyielding surface.

Crushup of the upper impact limiter occurs during the slap down phase; this phase (and the drop event itself) is terminated when the velocity of the upper impact end becomes zero.

The resulting center of mass vertical accelerations, which, for the cases examined, were less than 100 g's for all angles of impact, are then employed to estimate the resultant stresses on the fuel rods.

Fuel Rod Response

The load transfer paths from the cask to the fuel assemblies depend sensitively on the orientation of the cask that is dropped. For end-on drops, the load is transmitted axially through each fuel rod, from end plate to end plate. For side-on drops, on the other hand, the load transfer path to the fuel rods is through the basket to the spacer grids and end plates. Hence, whereas end drop responses can be modeled by considering only a single rod, side drop response analyses require a multiple rod assembly model.

Corner drop loadings can be treated through appropriate combinations of the end-on and side-on models.

The spacer grids play a very important role in side drops, and must be modeled in considerable detail. Unfortunately, the force transfer mechanism between the spacer grids and the fuel rods is highly nonlinear, primarily because of the flattening of the spacer contact springs and the buckling of the spacer grid frame.

In the side drop configuration, the spacer grids support the full weight of the rods. But as the load on the rods in increased, the spacer contact springs flatten out and lose their resistance. Consequently, the load is transferred to the spacer grid frame members until these reach their critical buckling load. After buckling, the spacer grids no longer support the fuel rods independently, and the fuel rods deflect further until they come to rest on top of each other and begin to participate in carrying the load. At this stage the claddings will ovalize until the pellets are contacted and they too begin to participate in the load transfer process.

Two types of rod elements are modeled for the side drop analyses. Tie rods (BWR) and control-rod guide tubes (PWR) are modeled with rigid attachments to the end plates of the assemblies, whereas the loose fuel rods, which are constrained to follow transverse and rotational motions of the end plates, are free to slide in and out of these plates. Furthermore, only the tie rods and control-rod guide tubes can transmit axial loads.

Under end drop loading conditions, the primary load is transmitted axially along the length of the fuel rods, thus a single rod model can be employed to define the assembly response. Moreover, this approach will necessarily maximize the response, since rod-to-rod interaction, caused by variations in deformation patterns among rods, would reduce the maximum lateral deformations which determine cladding failure for this loading condition. Extending this argument, it should thus be evident that lateral constrains on the fuel assembly play a critical role is determining cladding failure.

In the present analyses, the corner drop is separated into two events, and separate models are employed to characterize the two sets of loading conditions. The initial impact is modeled with the single rod model that is used for end-on impacts, but modified to accommodate lateral and rotational loading. Response in the slap down phase, on the other hand, is modeled using a modified version of the multiple rod assembly model.

Cladding Failure Criteria and Failure Modes

Once fuel rod response to the loading conditions has been determined, the analysis proceeds to an evaluation of the likelihood that the stresses experienced by the cladding will result in cladding failure. Thus, it is at this point that the treatment departs from one which is mechanistic in character to one of a probabilistic nature.

Fuel rod cladding failure can occur by ductile tearing of the material due to excessive strain, or by fracture at a pre-existing crack. For both types of failure, however, a probabilistic approach becomes warranted. In the former case, a probability function can be developed which relates the likelihood of ductile tearing due to strain on the magnitude of the force and on the ratio of hoop stress to axial stress.

For failure due to fracture, the probability of its occurrence is dependent upon the flaw density and its size distribution, and upon the fracture toughness of the material.

These two failure mechanisms provide three modes of failure of the cladding under shipping cask transport conditions. The first of these is transverse tearing of the cladding. This can result from tensile bending strains which could develop as the rods deform under load. In the case of a side drop, the bending strains are maximal at the spacer grids, whereas in an end drop, the strains are greatest at the attachment of the fuel rods to the bottom end-plate. The resultant failures are expected to be of a "pin-hole" type.

The second mode of failure is cladding fracture due to the extension of an existing flaw in the cladding. In the case of a side drop, this mode of failure can result from pinch loads due to the fuel rods stacking up on each other. The most likely region where this could occur is in the bottom rods at the basket supports, where the pinch loads would be maximal. Large lateral deflections of the fuel rods due to an end drop might also result in pinch loads sufficiently large to cause progation of an existing inner wall crack in the cladding. Cladding failures due to flaw extension will propagate longitudinally along the rod, resulting in a slit.

The third failure mode is an extension of transverse tearing sufficiently large to result in rod breakage.

Rod failures due to corner drop initial impact and slap down loads are similar to the end and side drop loadings. For the slap down event, the probability of failure would be highest at the secondary impact (slap down) end of the assembly.

Release of Radioactive Materials

Upon failure of the fuel rod cladding, radiotoxic materials are released by three different processes. the first of these involves the gaseous fission products; it is due to simple expansion of the gases in the interconnected void volume within the fuel rod into the shipping cask cavity. For the transport of 5-year-cooled fuel, the nuclides of concern are ^{85}Kr and ^{3}H.

The second mode of release involves radionuclides which are present within the rod in a chemical form that is volatile at temperatures of interest. The nuclides ^{134}Cs and ^{137}Cs dominate this mode of release, and the actual release mechanism is initially the purging action of the fill and fission product gases as they escape through the cladding defect. There is a longer term release mechanism for the volatile species as well; this mode of release, however, is by a gaseous diffusion process, and is unimportant for shipping cask safety assessment purposes.

The third mode of release concerns the ejection of fuel fines by the expanding gases at the time of cladding failure. These fines contain the same inventories of radioactive species as the bulk fuel, but may be enhanced in nuclides that are present in volatile form.

Models have been developed for each of the three release processes which are based upon experimental determinations of fission product releases from overheated fuel. In brief, these models indicate that only the material that resides within the interconnected voids within the fuel rods are available for release in a shipping cask environment. This is the so-called "gap fraction" of the total radionuclide inventory of the rod. The model for gaseous fission product escape permits complete release of the gaseous component of the gap fraction, but only a fraction of the volatile species, whose value is temperature dependent. Finally, 0.003% of the fuel rod contents are indicated to be released in the form of fuel fines. All three processes are assumed to be independent of size or location of the cladding failure, so long as complete fracture of the rod has not occurred.

Although impaction may create additional fuel fines, this will not appreciably alter the source term for fuel fines for the case of simple cladding breaches, but it may increase the extent of fines release into the shipping cask in the event of fuel rod breakage.

Once released into the cavity of the shipping cask, dilution of the radioactive materials occurs because of the much larger free volume of the shipping cask relative to that of a fuel rod, and the (assumed complete) mixing of the released material with the cask fill gas. In addition, attenuation of the fuel fines concentration will occur because of deposition and gravitational settling processes. Currently, insufficient information concerning the size distribution of the ejected fines is available to treat these processes in detail. Based upon observations of the behavior of fines released from overheated fuel, however, assuming that 10% of the ejected fines remains airborne sufficiently long that they remain available for release from the cask appears to be reasonably conservative.

RESULTS AND DISCUSSION

Because of the small concentrations of potentially releasable radioactive materials involved in shipping cask transportation environments, interactions among materials released can be ignored. Hence, for an estimate of the amounts released from cladding failures of fuel rods in a given assembly, the result is obtained by multiplying the calculated release from one of the rods by the total number of rods whose claddings fail. This approach thus permits considerations of the likelihood of cladding failure to be decoupled from the consequences of such failure occurring.

The Likelihood of Cladding Failure

Sample calculations have been performed using the models that have been briefly described previously to estimate the likelihood of fuel rod cladding failure involving the transport of a 15x15 PWR fuel assembly in a "generic" shipping cask. The results are summarized in Table I. The results are presented in terms of the likelihood of cladding failure of a rod as result of an event of the type noted in the table.

For the case of a 30-min fire, the probability of a rod containing a near-through-wall crack in its cladding was assumed to be one in ten thousand. For the remaining cases, the internal cladding wall crack size distribution (assumed to be due to pellet-cladding interaction) was assumed to be similar to that noted from an examination of North Sea offshore platform weldments [RODRIGUES 1980].

As can be seen from the data presented in Table I, fires appear to be inconsequential for the case examined (as stated previously) whereas side drop and near side drop events appear to have the greatest potential for causing fuel rod cladding failure. It is interesting to note, for purposes of comparison, that a 15 x 15 PWR assembly contains slightly more than 200 rods, and only about 10 PWR assemblies can be shipped at one time in the largest (rail) cask currently in use. Hence, the maximum number of rods involved in a given event is on the order of 2×10^3. The present analyses thus indicate a very low likelihood of multiple rod failures even under the accident conditions that are stipulated in 10CFR71.

TABLE I. Failure Probabilities Involving the Shipment of a 15 x 15 PWR Fuel Assembly in a Generic Shipping Cask

Assembly Loading Condition	Failure Probability Per Rod		
	Longitudinal Slit	Pinhole Rupture	Rod Breakage
30 ft. End Drop	1×10^{-9}	7×10^{-6}	8×10^{-7}
30 ft. Corner Drop (84° Drop Angle)	4×10^{-9}	9×10^{-6}	1×10^{-6}
30 ft. Corner Drop (2° Drop Angle)	5×10^{-5}	1×10^{-4}	2×10^{-5}
30 ft. Side Drop	4×10^{-5}	2×10^{-4}	5×10^{-5}
1 ft. Corner Drop (60° Drop Angle)	1×10^{-5}	1×10^{-6}	2×10^{-7}
Fire[a]	1×10^{-11}	-	-

[a]Based upon an incipient crack in the cladding with a probability of 1×10^{-4}.

The Consequences of Cladding Failure

An irradiation history identical with that actually experienced by a 15 x 15 PWR fuel assembly was selected, and estimates of the inventory were made using the ORIGEN2 computational program [CROFF 1980, CROFF 1983]. This inventory and the operational history were then employed to calculate the gap fractions of pertinent nuclides. For this purpose the methodology described in the standard ANSI/ANS-5.4-1982 [ANSI 1982] was employed to estimate the gap fraction for ^{85}Kr, and the corresponding gap fractions for ^{124}Cs and ^{137}Cs were taken to be identical to that of ^{85}Kr. The gap fraction for ^{3}H was arbitrarily (and conservatively) taken to be 0.5.

All of the ^{85}Kr and ^{3}H in the fuel-cladding gap region and plenum were assumed to be released at the time of cladding failure, along with 0.003% of the fuel as fuel fines. In addition to the cesium (and krypton) contained in the fuel fines, volatile cesium species which are purged from the gap regions were estimated using a model developed for cesium release from overheated fuel [LORENZ 1979]. The resulting concentrations within a shipping cask of the same free volume ($0.155 m^2$) as the NLI-1/2 shipping cask is presented in Table II. (The results have been reduced to a common radiotoxicity basis by expressing the concentrations as multiples of the respective A_2 values as presented in 10CFR71 [CFR 1987]. In addition, a "default value," $A_2 = 0.4$ Ci, was used to account for all other radionuclides not specifically shown in the table.)

It is clear from an examination of the values given in Table II that the actinides, released as fuel fines, constitute the main radiologic hazard from material released from rods whose claddings fail in the course of transport. Also, failure of a single fuel rod cladding results in a concentration of 0.161 A_2 Ci/m^3 in an NLI-1/2 shipping cask. This constitutes the concentration C_i (where i = N for normal and i = A for accident conditions) which relates the release rate R_i with the leak rate L_i for safety assessment purposes,

$$L_i = R_i/C_i. \quad (1)$$

Release rate criteria specified in 10 CFR 71 are, for normal conditions of transport,

$$R_N = 1 \times 10^{-6} A_2 \text{ Ci/h}, \quad (2)$$

and, for accident conditions,

$$R_A = A_2 \text{ Ci/week}. \quad (3)$$

If the value cited above for C_i is employed, these criteria yield leak rates of 1.73×10^{-3} cm^3/s for normal conditions, and $10.3/cm^3/s$ for accident conditions of transport.

The inadvertent use of air as the cask fill gas can alter this result appreciably, but air inleakage into a cask that would normally meet the release rate criteria can be shown to have only a minor effect on the results presented in the sample calculation presented here.

TABLE II. Radionuclide Concentrations in an NLI-1/2 Shipping Cask Resulting from the Failure of a Single PWR Fuel Rod[a]

Nuclide	A_2(Ci)	Concentration (Ci/m^2)[b]
^{3}H[c]	1000	$3.29\times10^{-3}A_2$
^{85}Kr[d]	1000	$1.01\times10^{-3}A_2$
^{90}Sr	0.4	$7.50\times10^{-3}A_2$
^{90}Y	10	$3.00\times10^{-4}A_2$
^{106}Ru	7	$9.07\times10^{-5}A_2$
^{125m}Te	100	$4.14\times10^{-7}A_2$
^{134}Cs[e]	10	$2.18\times10^{-3}A_2$
^{137}Cs[e]	10	$3.57\times10^{-3}A_2$
^{144}Ce	7	$5.31\times10^{-5}A_2$
^{147}Pm	25	$5.29\times10^{-5}A_2$
^{235}U	0.2	$2.82\times10^{-9}A_2$
^{238}Pu	0.003	$5.44\times10^{-2}A_2$
^{239}Pu	0.002	$6.81\times10^{-3}A_2$
^{240}Pu	0.002	$1.25\times10^{-2}A_2$
^{241}Pu	0.1	$4.82\times10^{-2}A_2$
^{241}Am	0.008	$6.56\times10^{-3}A_2$
^{244}Cm	0.01	$1.34\times10^{-2}A_2$
Others	0.4	$1.45\times10^{-3}A_2$
Total		$1.61\times10^{-1}A_2$

[a]Oconee-1 Rod 08639 irradiated to 38.2 MWD/kgU after 5 years' decay.
[b]Only 10% of the fuel fines released are assumed to remain airborne.
[c]Gap inventory and fuel fines, but only 10% assumed to remain with fuel fines.
[d]Gap inventory and fuel fines.
[e]Gap purge and fuel fines.

REFERENCES

CFR 1987. United States Code of Federal Regulations, Title 10, Part 71 (1987).

DUFFEY 1988. T. A. Duffey, S. H. Sutherland, and G. C. Williams, "Characterization of Spent Fuel Transportation Environments," APTEK Report A-88-14R (Rev. 1) (1988).

BURIAN 1985. Burian, R. J., Kok, K. D., DiSalvo, R., Balmert, M. E., Freeman-Kelly, R., and Pentinan, A. W., "Response of Spent LWR Fuel to Extreme Temperatures," Battelle Columbus Laboratories Report SAN 85-7213 (1985).

RODRIGUES 1980. Rodrigues, P.E.L.B., Wong, W. K., and Rogerson, J. H., "Weld Defect Distributions in Offshore Platforms and Their Relevance to Reliability Studies, Quality Control, and In-Service Inspection," Cranfield Institute of Technology, Offshore Technology Conference (1980).

CROFF 1980. Croff, A. G., "ORIGEN2 - A revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code," Oak Ridge National Laboratory Report ORNL-5621 (1980).

CROFF 1983. Croff, A. G., "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," Nuc. Tech. 62, 335 (1983).

ANSI 1982. "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," ANSI/ANS-5.4-1982, American Nuclear Society (1982).

LORENZ 1979. Lorenz, R. A., Collins, J. L., and Malinauskas, A. P., "Fission Product Source Terms for the Light Water Reactor Loss-of-Coolant Accident," Nuc. Tech. 46, 404 (1979).