Bounding Estimates of Criticality Effects of Unburned LWR Fuel Assembly Tips for Casks Assuming Burnup Credit

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INTRODUCTION AND CONCLUSIONS

The criticality scenario treated here of a large, (36 PWR Assemblies) dry, spent fuel cask is based on the puncturing and subsequent submersion of the cask into water which enters the cask at some reasonable rate. All assembly tips (one foot length, top and bottom) are assumed to be relatively unburned (2w/o enriched in U-235), whereas, the middle ten feet of each assembly has a residual discharge U-235 enrichment of 0.8w/o. The 36 PWR assemblies reside contiguously in the cask cavity without structural baskets or fixed neutron poisons. This, together with an enrichment of 0.8w/o for over 83% of the active fuel length, represent the factors for burnup credit in transportation. The tip enrichment with the in-leakage water are the essential ingredients necessary for super-criticality to occur.

A comparison with the super-critical SPERT-OXIDE reactor experiments shows that the experiment and the cask-contents accident-environment are equivalent on a nuclear-criticality basis. Justification using this similarity argument is given in subsequent sections of this paper. Dollar step insertion of reactivity, initial reactor period, maximum energy released (mw-sec and total fission yield) from the nuclear excursion, maximum pressure, number of fuel rods ruptured and eventual shutdown mechanisms are all estimated for the cask directly from SPERT-OXIDE data. The SPERT experiments were performed in an open tank. The cask-water system is likewise considered an open tank. A step insertion of reactivity can be realized in the instance when a submerged cask is suddenly uprighted (recovery operations) allowing the water on the cask insides to quickly rush to the cask bottom - covering the assembly tips containing the higher U-235 enrichment.

The pertinent system parameters and general results are given in TABLE 1. The SPERT-OXIDE lattice is compared with a typical PWR lattice as cask contents in TABLE 2. All SPERT data was taken from Nyer et al. 1964, Spano 1963, and Thompson and Beckerley 1964.

The major conclusions of the Spent Fuel Cask criticality scenario are:

1. A valid comparison under super-prompt criticality conditions can be made between the 36 PWR spent cask and the SPERT-OXIDE core excursion experiments. Thus, predictions of criticality consequences for the cask, by proper scaling of the SPERT results is possible. For the cask criticality they were:

   a. Maximum energy release: 50mw-sec
   b. Total fission yield: 1.5×10^{18}

c. Number rods burst: 24 (out of 7344)
d. Peak pressure: ≤ 10 atm.
e. Fuel temperature: ≤ 1800°C
f. Shutdown mechanism: prompt negative U-238 Doppler effect.
g. Physical damage: assuming the fuel-water region of cask to be an open system: lid remains securely on cask; no bowing of fuel rods; no damage of any consequence expected.
h. Direct dose:
for cask in air: lethal dose (500REM) at 33 feet; acceptable emergency (25REM) at 164 feet; one REM at 590 feet.
for cask in water: above distances are reduced by factor of 2.5.
i. Cloud doses:
for cask in air:

Whole body immersion dose vs distance		Thyroid dose (0-8 hr) vs distance	
10 meters:	650 REM	10 meters:	2000 REM
100 meters:	60 REM	100 meters:	140 REM
1000 meters:	1.4 REM	1000 meters:	3.0 REM

2. For slow, ramp insertions of reactivity (water gradually filling the fuel assembly tip region) would lead to a criticality of much lower power than the assumed step insertion. In this case increase of the moderator temperature would decrease its density leading eventually to a shutdown of the chain reaction. A possibility does exist for this system to oscillate from subcritical to critical for an unknown period, since cold water can be furnished indefinitely. The record supercritical-critical system was the Hanford plutonium solution which lasted some 37 hours before shutting down. The solid system of PWR assemblies is not as sensitive to geometric and material changes and fluctuations as solutions. The total fission yield is expected to be lower than that from the step excursion.

3. The burst spent fuel rods constitute a river contamination problem. High Cs-137 curie levels will be dissolved in the cask water eventually flowing into the river proper. This probably is the most serious radiological impact to the general public from the cask criticality.

ANALYSIS AND ASSUMPTIONS

Reactor endurance (burnup) calculations, using current fuel management schemes, predict that LWR fuel assembly tips (8 inches to 12 inches at top and bottom of assemblies) are not as fully burned as the central (about 90% of assembly) portion of the fuel assembly. The tip region can have as much as 50% to 125% more U-235 residual enrichment than that indicated from the quoted average discharge enrichment. For example, for an average discharge enrichment of 0.9 w/o U-235, the tip regions can have an effective U-235 enrichment of about 2.0 w/o. It is assumed that the tip regions do not contain any built-up fissile plutonium or (since fissioning was minimal in this tip region) fission products.

Although these enrichment "end-effects" may be an artifact of a mathematical solution to a boundary-value problem (i.e., flux or power unrealistically plummeting in the axial directions to a near zero value) it is supported by axial gamma scans of actual spent fuel assemblies and there is no experimental basis to disregard this potential end-effect at this time. Since enrichments 1 w/o U-235 and greater can become critical with water moderation, these tip regions can pose a criticality potential for spent fuel casks not containing fixed neutron poisons when flooded.

To estimate the criticality potential of tips of PWR fuel assemblies a number of KENO-Monte Carlo criticality problems were performed for 1, 4, 9, 16, 25, and 36 close-fitting PWR fuel assemblies, one foot in height, moderated and reflected by water, as a function of U-235 enrichment. Results of keff as a function of enrichment are given in Figure 1. This figure shows that an infinite number of fuel assembly tips, closely fit, cannot go critical for U-235 enrichments less than 1.55 w/o. For 36 fuel assemblies at 2 w/o enrichment, the tips when moderated reflected by water would have a keff of 1.031.

The method used here for the criticality analysis of the spent fuel cask is one of direct comparison with the relevant SPERT oxide experiments. Examination of the lattice parameters of the cask contents with the SPERT oxide core of Table 2 shows general overall nuclear similarities. The difference of enrichment (2 w/o vs 4 w/o of SPERT), W/F (1.73 vs 1.58 of SPERT), and cladding material (Zr vs SS of SPERT) are not expected to significantly change the neutron spectrum characteristics of the respective systems. The higher enrichment of SPERT together with its lower moderating power is countered by the cask's contents lower enrichment together with its stronger moderating power. The most fundamental parameter (or process) that should be the same in comparing two super-critical systems is the physical approach and amount of excess reactivity (making the system supercritical) and the manner in which it is added to each system.

The SPERT core was made prompt super-critical by the sudden withdrawal of the single transient control rod located at the geometric center of the core. The worth of the step insertion of reactivity added to the SPERT core by this sudden loss of central absorber was measured to be $3.30--or the keff of SPERT at the end of the withdrawal was 1 + (3.30) (.0065) = 1.021. From Figure 1 we note that a system of 36 PWR tips (1 foot tall) fully moderated and water reflected at an enrichment of 1.95 w/o U-235, has a keff of 1.021. To effect a step insertion of reactivity into the spent fuel cask it is assumed that the cask is punctured and subsequently submerged on its side into river water where water accumulates in the side portion of the cylinder between the cylindrical walls up to the dry longitudinal plane bounding the 6 x 6 fuel assembly matrix. The fuel is dry and the water stops accumulating. The cask is then suddenly uprighted as a right circular cylinder in which the water rushes to the base of the cylinder mixing with the higher enriched tips bringing keff to 1.021. The water-fall in the cask is assumed to have the same speed of the spent rod withdrawal and thus arrive at the same insertion of $3.30 as SPERT.

The assumption is made that the absorption spectrum perturbation by withdrawal of an absorber rod in SPERT is equivalent to the slowing down spectrum perturbation by introducing water in the cask. Since the SPERT excursion is obviously due to a faster step insertion and in the core region of highest Importance applying SPERT results to the cask is extremely conservative since there is no corresponding spatial Importance in the cask.

The conclusion from the above super-criticality equivalence is that the consequences from the SPERT-oxide experiment, including the shutdown mechanism, are directly applicable to the spent fuel cask by a simple scaling procedure.

Step insertions of reactivity (for the same amount of reactivity) are much more efficient in putting a critical system on a short period excursion and releasing more energy than the equivalent ramp insertion of reactivity.

For example in the SPERT oxide core, 2$ of reactivity inserted in a step fashion put the core on a 3.2 ms period. In order to arrive at the same period, a ramp insertion of 8$/sec was required--for a sufficient time--to inject at least 2$ reactivity into the system.

Thus only a step insertion is considered for the cask criticality analysis and the rate of water-fill into the cask does not pertain since no ramp insertions are assumed in this study.

SPERT CORE AND CALCULATION OF PARAMETERS

The SPERT-oxide experiments were devised for the investigation of the dynamic behavior of water-moderated, low-enriched, UO_2 fuel and the relative importance of various reactivity feedback effects on the kinetics and inherent safety of such reactors by means of step-initiated self-limiting power excursions. These tests, for the first time, demonstrated the effectiveness of the Doppler shutdown mechanism of U-238 in low-enrichment reactors.

The SPERT-Oxide core (Spano, 1963) was composed of 592 B&W N.S. Savannah $U(4)0_2$ rods, 6 ft. long, welded-seam 304 ss clad. The UO_2 had a density 87% theoretical. These rods, placed on a lattice (Table 2 specs) in an open 4 foot diameter water tank with no provision for pressurization or forced coolant flow. The octogonal-shaped core was divided by 4 control blades (each 7.5 inches wide), which acted as a bank. The central portion of this cruciform had a water gap of 4.5 inches wide along each axis separating the pairs of control blades. The single transient cruciform control rod (2.25 inches wide) resided in the central position. It acted separately from the bank of 4 control rods. Transients were generally accomplished by achieving a critical system with the bank of 4 control blades and the transient inserted not necessarily at the same depth. Sudden withdrawal of the single transient rod then was effected placing the critical system on a short period. The prompt "α" (described below) and the energy release were then measured.

An observable of major importance in the kinetic analysis of critical systems made supercritical by a step insertion of reactivity is the prompt "alpha": α, which is equal to the reciprocal of the exponential period, τ, induced by the injected positive reactivity. It can be represented as the logarithmic derivative of the power:

$$\frac{1}{p}\,\frac{dp}{dt} = \alpha$$

giving: $p = p(o)e^{\alpha t}$

a step insertion of reactivity specified by the relation

$$\alpha = \frac{\$}{(l/\beta)}$$

Where \$ represents the instantaneous (step) degree of super-prompt criticality in dollar units, which puts the system on an initial period τ; l is the prompt neutron lifetime and β is the delayed neutron fraction, taken as 0.0065.

The fastest transient (shortest period) experienced by the SPERT-oxide core (Nyer et al., 1964) was accomplished by a step insertion of \$3.30 reactivity measuring an α of 645 sec^{-1}. This translated to an initial period of 1.55 ms and generating a total energy release of 165 MW-sec. Using $3.12x10^{10}$ fissions equal to 1 watt-sec, this total energy release translates to $5.1x10^{18}$ fissions as the yield in this excursion.

Measurement of $\alpha = (\beta/\ )\ \$$ gives the prompt neutron lifetime for the given transient with β taken as 0.0065. Thus

$$l \text{ prompt} = \frac{(0.0065)}{654 \text{ sec}^{-1}}\ (3.30) = 3.3 \text{ x } 10\text{-}5 \text{ sec}$$

Thompson and Beckerley (1964) p. 675 gives a plot of measured total energy release as a function of initial reactor period for a series of reactor excursions (BORAX's SPERT's and SL-1). This plot is given in this paper as Figure 2 showing a 1.55 ms period resulting in about 165 MW-sec (when the SPERT curve is extended) agreeing with values reported elsewhere.

The cask contents -36 PWR tips at a ~1.95 w/o U-35 enrichment fully water-moderated and reflected was calculated by KENO-IV 27 groups to have the same k prompt of 1.021 (as did SPERT) and an average prompt neutron lifetime of $5.5x10^{-5}$ sec.

This transforms to a prompt α of $\frac{.0065}{5.5x10^{-5} \text{ sec}}$ (3.30) = 390 sec^{-1}

with a corresponding initial period of 2.56 ms. Using Figure 2, for this period a total energy release of 50mw-sec results or equivalent a total yield of $1.5x10^{18}$ fissions.

This $1.5x10^{18}$ fissions forms the basis for the radiological impact to the general public using standard accident-dose versus fission yield curves.

REFERENCES

Nyer, W.E., Bright, G.O., & McWhorter, R.J., Reactor Excursion Behavior p/283 Proc. U.N. Intern. Conf. Peaceful Uses At.Energy, 3rd Geneva, 13: 13-21 (1964).

Spano, A.H., "Self-Limiting Power Excursion Tests of Water-Moderated Low-Enrichment UO_2 Core", Nuclear Science & Engineering, 15, 37-51, (1963).

Thompson, T.J. & Beckerley, J.G. (Editors), The Technology of Nuclear Reactor Safety. Vol 1, Reactor Physics and Control MIT Press, 1964.

TABLE 1

COMPARISON OF EXCURSION PARAMETERS
AND EXPECTED RESULTS

PARAMETER	SPERT-OXIDE CORE (Observed or Inferred)	CASK CONTENTS (Expected)
Reactivity ($)	3.30; $k_p = 1.021$	$k_p = 1.021$
Step Insertion effected by:	Sudden withdrawal of transient control rod.	Sudden uprighting of submerged cask on side.
α (sec^{-1})	645	390
Initial period (ms)	1.55	2.56
Prompt neutron lifetime ℓ (sec)	3.3×10^{-5}	5.5×10^{-5}
Tot Energy Release (Mw-sec)	165	50
Tot Fission Yield (fissions)	5×10^{18}	1.5×10^{18}

TABLE 2

LATTICE PARAMETERS

PARAMETER	SPERT-OXIDE CORE	SPENT FUEL CASK CONTENTS
Number PWR Assy's/Rods	3/592	36/7344
Fuel Composition	UO_2	UO_2
U-235 Enrichment w/o	4.0	1.95
Rod Diameter (in)	0.444	0.366
Clad/Thickness (in)	SS/0.028	Zr/0.024
Active Rod Length (ft)	6.0	1.0
Rectangular Pitch (in)	0.663	0.563
Volume Frac, w/f	1.575	1.731

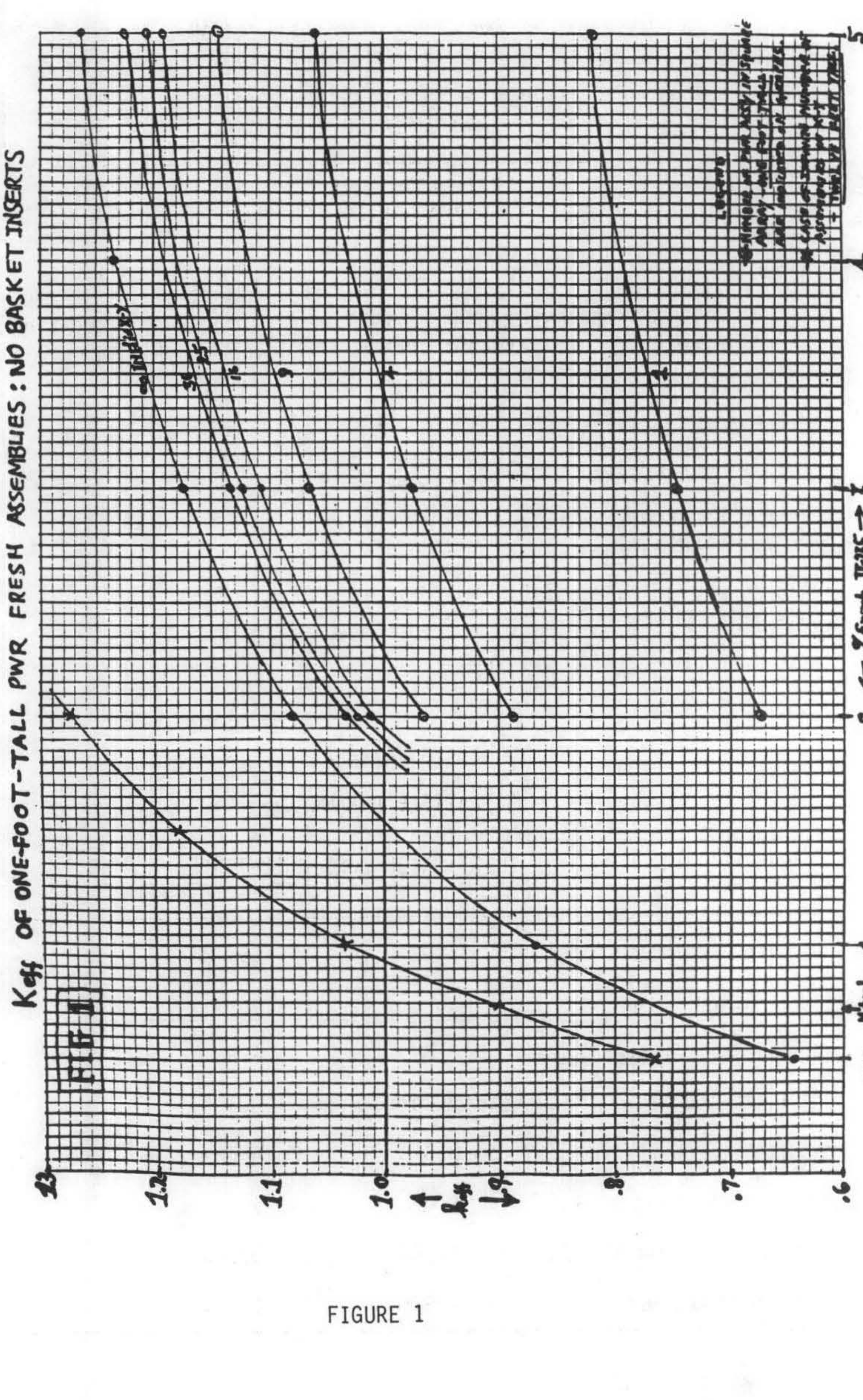
Keff OF ONE-FOOT-TALL PWR FRESH ASSEMBLIES : NO BASKET INSERTS
FIG 1
% Enrich. U-235
Natural Uranium
0.5
1
2
3
4
5
1.3
1.2
1.1
1.0
.9
.8
.7
.6
keff
36
25
16
9
4
2

FIGURE 1

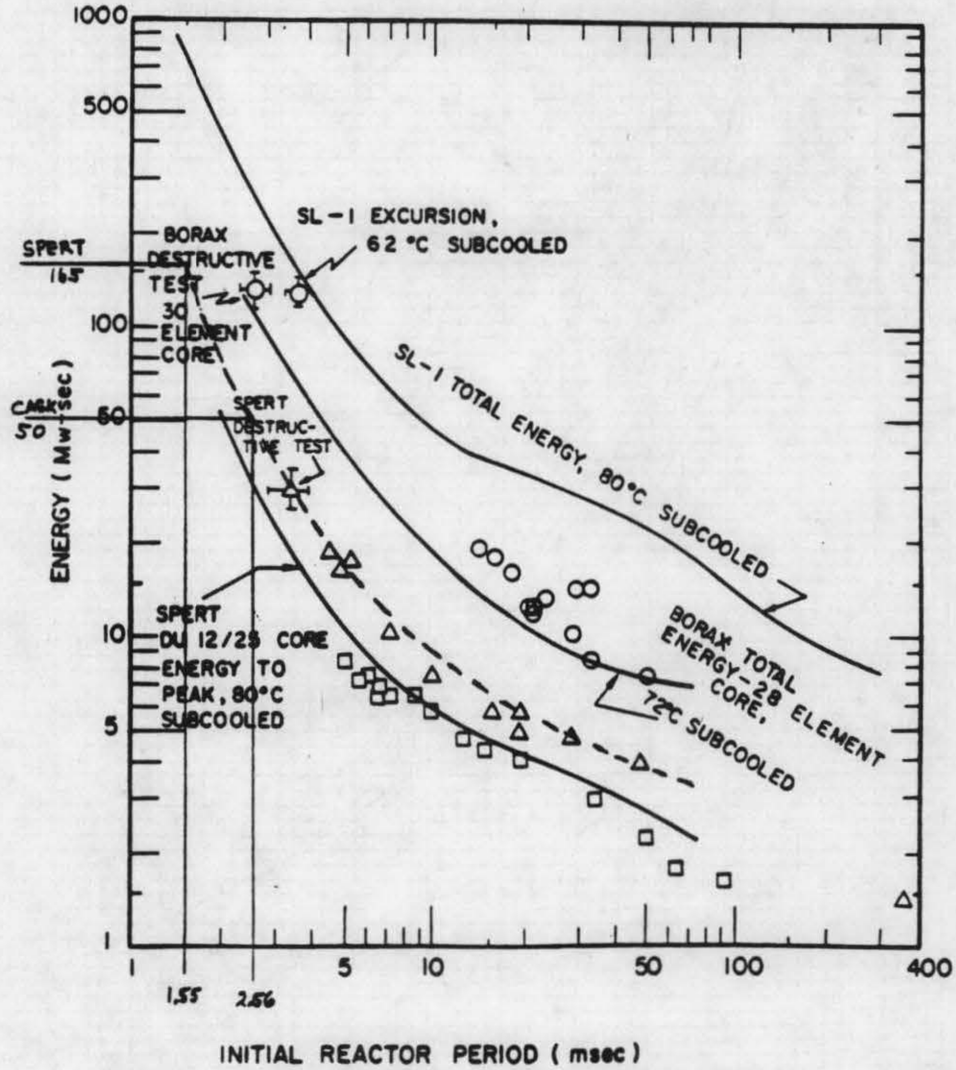
SL-1 EXCURSION, 62 °C SUBCOOLED
BORAX DESTRUCTIVE TEST 30 ELEMENT CORE
SPERT 165
CASK 50
SPERT DESTRUCTIVE TEST
SL-1 TOTAL ENERGY, 80°C SUBCOOLED
BORAX TOTAL ENERGY-28 ELEMENT CORE, 72°C SUBCOOLED
SPERT DU 12/25 CORE ENERGY TO PEAK, 80°C SUBCOOLED
ENERGY (Mw-sec)
INITIAL REACTOR PERIOD (msec)
1.55
2.56
1000
500
100
50
10
5
1
1
5
10
50
100
400

FIGURE 2