December 14, 1999

MEMORANDUM TO: John Hannon, Chief
Plant Systems Branch
Division of Systems Safety and Analysis

FROM: Jared S. Wermiel, Chief /s/
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: DRAFT CRITICALITY ASSESSMENT OF CRITICALITY EVENTS IN DECOMMISSIONED SPENT FUEL POOLS

We have completed our assessment of the possibility of a criticality accident in a decommissioned spent fuel pool. We have identified a compression event of a low density, unpoisoned storage rack as an event that could lead to an unplanned criticality if compensatory measures are not taken. High density, poisoned racks would not cause criticality concerns. We have also identified two compensatory measures. Some combination of these will be necessary to provide reasonable assurance that a compression event of a low density rack will not lead to a criticality accident.

1. The most reactive assemblies (most likely the fuel from the final cycle of operation) could be scattered throughout the pool; and/or

2. all storage pools, regardless of reactor type, could be borated

Attachments:
As stated

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Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools

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1 Introduction

The staff has performed a series of calculations to assess the potential for a criticality accident in the spent fuel pool of a shutdown nuclear power plant. This work was undertaken to support the staff's efforts to develop a decommissioning rule. Unlike operating spent fuel storage pools, decommissioned pools will have to store a large number of highly reactive assemblies for extended periods of time which were used in the final operating cycle of the reactor. Operating reactors typically only store highly reactive assemblies for short periods of time during refueling or maintenance outages. As we will see in this report, the loss of geometry alone could cause a criticality accident unless some mitigative measures are in place.

When spent fuel pools were originally conceived, they were intended to provide short term storage for a relatively small number of assemblies while they decayed for a period of time sufficient to allow their transport to a long term storage facility. Because a long term storage facility is not available, many reactor owners have had to change the configuration of their spent fuel pools on one or, in some cases, several occasions. This practice has led to a situation where there are many different storage configurations at U.S. plants utilizing some combination of geometry, burnup, fixed poisons, and so on, to safely store spent fuel.

The current state of spent fuel pools significantly complicates the task of generically analyzing all potential spent fuel pool configurations. Therefore, the staff decided to take a more phenomenalogical approach to the analysis. Rather than trying to develop specific scenarios for all of the different types of loading configurations, we decided to analyze storage rack deformation and degradation using typical storage racks. The results of these calculations will be used to formulate a set of generic conclusions regarding the physical controls necessary to prevent criticality. The impact of this assumption on the conclusions in this report will be discussed throughout the text. Furthermore, for the purposes of this work, it is assumed that the event is unrecoverable when the water level reaches the top of the fuel. This means that we will not consider events such as a loss of water leading to a low density optimal moderation condition caused by firefighting equipment.

It is important to reinforce the point that these analyses are intended as a guide only and will be used to evaluate what controls are either currently in place or will need to be added to ensure that criticality will be prevented. These analyses will not be used to develop specific numerical limits which must be in place to control criticality as they cannot consider all of the possible plant specific variables. We will, however, define the controls that are necessary either individually or in combination to prevent a criticality accident.

2 Description of Methods

The criticality analyses were performed with three-dimensional Monte Carlo methods using ENDF/B-V based problem specific cross sections (Ref. 1). Isotopic inventories were predicted using both one- and two-dimensional transport theory based methods with point depletion. SCALE 4.3 (Ref. 2) was used to perform the Monte Carlo, one-dimensional transport, cross section processing, and depletion calculations. Specifically, the staff used KENO-VI, NITAWL-II, BONAMI, XSDRN, and ORIGEN. The two-dimensional transport theory code NEWT (Ref. 3) was used for Boiling Water Reactor (BWR) lattice depletion studies. NEWT uses the method of characteristics to exactly represent the two-dimensional geometry of the problem. NEWT uses ORIGEN for depletion. Cross section data were tracked and used on a pin cell basis for the BWR assessments. The staff developed post processing codes to extract the information from NEWT and create an input file suitable for use with SCALE. Both the 238 and the 44 group ENDF/B-V based libraries were used in the project. Refer to Appendix A for a listing of one of the input decks used in this analysis. SCALE has been extensively validated for these types of assessments. (see References 4, 5, and 6)

3 Problem Definition

Compression (or expansion) events were analyzed in two ways. First, the assembly was assumed to crush equally in the x and y directions (horizontal plane). Analyses were performed with and without the fixed absorber panels *without* soluble boron and with fuel at the most reactive point allowed for the configuration. In these cases, the fuel pin pitch was altered to change the fuel to moderator ratio. These scenarios are intended to simulate the crushing (or expansion) of a high density configuration when little or no rack deformation is necessary to apply force to the fuel assembly. The scenarios are also applicable to low density rack deformation in which the rack structure collapses to the point at which force is applied to the assemblies. The second type of compression event involved changing the intra-assembly spacing, but leaving the basic lattice geometry unchanged. These simulations were intended to simulate compression events in which the force applied to the rack is insufficient to compress the assembly.

4 Discussion of Results

Several observations are common to both Pressurized Water Reactor (PWR) and BWR rack designs. First of all, poisoned racks should remain subcritical during all compression type events assuming that the poison sheeting remains in place (in other words, that it compresses with the rack and does not have some sort of brittle failure). Secondly, criticality cannot be precluded by design following a compression event for low density, unpoisoned (referring to both soluble and fixed poisons) storage.

4.1 PWR Spent Fuel Storage Racks

The analyses and this discussion will differentiate between high and low density storage. High density storage is defined as racks that rely on both fixed poison sheets and geometry to control reactivity and low density storage relies solely upon geometry for reactivity control. The results of the analyses for the high density storage racks is summarized in Figure 1. When discussing Figure 1 the reader should keep in mind that the analyses supporting Figure 1 were performed without soluble boron and with fuel at the most reactive point allowed for the rack. These assumptions represent a significant conservatism of at least 20 percent delta-k. Figure 1 demonstrates that even with compression to an optimal geometric configuration, criticality is prevented by design (for these scenarios we are not trying to maintain a k_{eff} less than 0.95). The poison sheeting, boral in this case, is sufficient to keep the configuration subcritical.

The results for the low density storage rack are given in Figure 2. As can be seen, criticality cannot be entirely ruled out on the basis of geometry alone. Therefore, one must then examine the conservatism implicit in the methodology and assess whether there is enough margin to not require any additional measures for criticality control. There are two main sources of conservatism in the analyses; using fuel at the most reactive state allowed for the configuration and not crediting soluble boron. By relaxing the assumption that all of the fuel is at its peak expected reactivity, we have demonstrated by analyzing several sample storage configurations that the rack eigenvalue can be reduced to approximately 0.998 (see Table 1). The storage configurations analyzed included placing a most reactive bundle every second, fourth, sixth and eighth storage cell (see Figure 3). The assemblies used between the most reactive assembly were defined by burning the 5 w/o U_{235} enriched Westinghouse 15x15 assembly to 55 GWD/MTU which is a typical discharge burnup for an assembly of this type. This study did not examine all possible configurations so this value should be taken as an estimate only. However, the study does suggest that scattering the most reactive fuel throughout the pool will substantially reduce the risk of a criticality accident. It is difficult to entirely relax the assumption of no soluble boron in the pool, but its presence will allow time for recovery actions during an event that breaches the liner and compresses the rack but does not rapidly drain the pool.

Although not all-inclusive because all fuel and rack types were not explicitly considered, the physical controls that were identified should be generically applicable. The fuel used in this study is a Westinghouse 15x15 assembly enriched to 5 w/o U_{235} with no burnable absorbers. The Westinghouse 15x15 assembly has been shown by others

(Ref. 7) to be the most reactive PWR fuel type when compared to a large number of different types of PWR fuel. Furthermore, the use of 5 w/o U_{235} enriched fuel will bound all available fuel types because it represents the maximum allowed enrichment for commercial nuclear fuel.

4.2 BWR Spent Fuel Storage Racks

In these analyses, we will also differentiate between high and low density BWR racks. The conservatism inherent in the analyses must be considered (for BWR racks, the use of the most reactive fuel allowed only) when considering the discussion of these results. The results of the analyses of high density BWR racks are given in Figure 4. As can be seen, criticality is prevented by design for the high density configurations. The poison sheets remain reasonably intact following the postulated compression event. The poison sheeting (in this case Boraflex) is sufficient to maintain subcriticality.

The results of the low density BWR rack analyses are shown in Figure 5. Here, as with the PWR low density racks, criticality cannot be prevented by design. Once again we will assess the impact of eliminating some of the conservatism in the analyses which in the case of BWR storage is only related to the reactivity of the assembly. Analyses were performed placing a most reactive assembly in every second, fourth, sixth and eighth storage cell. The assemblies placed between the most reactive assemblies were defined by burning the 4.12 w/o enriched General Electric (GE) 12 assembly to 50 GWd/MTU These analyses demonstrate that it is possible to reduce the rack eigenvalue to approximately 1.009 (see Table 1). As previously mentioned, this study did not study all possible configurations so this value should be taken as an estimate only. Because BWR pools are not borated, there is no conservatism from the assumption of no soluble boron.

Boraflex degradation is another problem that is somewhat unique to BWR spent fuel storage racks. This is true because of the fact that BWR storage pools do not contain soluble boron that provides the negative reactivity in PWR pools to offset the positive effect of Boraflex degradation. Therefore, some compensatory measures need to be in place to provide adequate assurance that Boraflex degradation will not lead to a criticality event. In operating reactor spent fuel pools that use Boraflex, licensees use some sort of surveillance program to ensure that the 5 percent subcritical margin is maintained. These programs should be continued during and following decommissioning. No criticality calculations were performed for this study to assess Boraflex degradation because it is assumed that the loss of a substantial amount of Boraflex will most likely lead to a criticality accident.

These analyses are not all inclusive, but we believe that the physical controls identified are generically applicable. We examined all of the available GE designed BWR assemblies for which we had information and identified the assembly used in the study to have the largest K_{inf} in the standard cold core geometry (in other words, in the core with no control rods inserted at ambient temperature) at the time of peak reactivity. This assembly was a GE12 design (10x10 lattice) enriched to an average value of 4.12 w/o U_{235}. Only the dominant part of the lattice was analyzed and it was assumed to span the entire length of the assembly. This conservatism plus the fact that the assembly itself is highly enriched and designed for high burnup operation has led the staff to conclude that these analyses can be generically applied to BWR spent fuel storage pools.

5 Recommendations and Conclusions

One scenario that has been identified which could lead to a criticality event is a heavy load drop that compresses a low density rack filled with fuel at its peak expected reactivity. This event is somewhat unique to decommissioned reactors because there are many low burnup (high reactivity) assemblies stored in the spent fuel pool that were removed from the core following its last cycle of operation.

To mitigate the consequences of the compression of a low density rack, we have formulated two recommendations. These should be used individually or in combination.

First, the most reactive assemblies (most likely the fuel from the final cycle of operation) could be scattered throughout the pool, or placed in high density storage if available. Second, all storage pools, regardless of reactor type, could be borated.

6 References

1. "ENDF/B-V Nuclear Data Guidebook," EPRI-NP 2510, July 1982.

2. "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200. Oak Ridge National Laboratory, 1995.

3. Tony Ulses, "Evaluation of NEWT for Lattice Physics Applications," Letter Report, May 1999.

4. M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross Section Library for use in Criticality Safety Analysis," NUREG/CR-6102, Oak Ridge National Laboratory, 1994.

5. O.W. Hermann, et. al., "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.

6. W.C. Jordan, et. al., "Validation of KENOV.a Comparison with Critical Experiments," ORNL/CSD/TM-238, Oak Ridge National Laboratory, Oak Ridge National Laboratory, 1986.

7. "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools C and D," HI-971760, Holtec International, May 26, 1998, (Holtec International Proprietary)

Appendix A

Sample Input Deck Listing

```
=csas26   parm=size=10000000
KENO-VI Input for Storage Cell Calc. High Density Poisoned Rack
238groupndf5  latticecell
'Data From SAS2H - Burned 5 w/o Fuel
o-16      1 0 0.4646E-01 300.00 end
kr-83     1 0 0.3694E-05 300.00 end
rh-103    1 0 0.2639E-04 300.00 end
rh-105    1 0 0.6651E-07 300.00 end
ag-109    1 0 0.4459E-05 300.00 end
xe-131    1 0 0.2215E-04 300.00 end
'xe-135   1 0 0.9315E-08 300.00 end
cs-133    1 0 0.5911E-04 300.00 end
cs-134    1 0 0.5951E-05 300.00 end
cs-135    1 0 0.2129E-04 300.00 end
ba-140    1 0 0.1097E-05 300.00 end
la-140    1 0 0.1485E-06 300.00 end
nd-143    1 0 0.4070E-04 300.00 end
nd-145    1 0 0.3325E-04 300.00 end
pm-147    1 0 0.8045E-05 300.00 end
pm-148    1 0 0.4711E-07 300.00 end
pm-148    1 0 0.6040E-07 300.00 end
pm-149    1 0 0.6407E-07 300.00 end
sm-147    1 0 0.3349E-05 300.00 end
sm-149    1 0 0.1276E-06 300.00 end
sm-150    1 0 0.1409E-04 300.00 end
sm-151    1 0 0.7151E-06 300.00 end
sm-152    1 0 0.5350E-05 300.00 end
eu-153    1 0 0.4698E-05 300.00 end
eu-154    1 0 0.1710E-05 300.00 end
eu-155    1 0 0.6732E-06 300.00 end
gd-154    1 0 0.1215E-06 300.00 end
gd-155    1 0 0.5101E-08 300.00 end
gd-156    1 0 0.2252E-05 300.00 end
gd-157    1 0 0.3928E-08 300.00 end
gd-158    1 0 0.6153E-06 300.00 end
gd-160    1 0 0.3549E-07 300.00 end
u-234     1 0 0.6189E-07 300.00 end
u-235     1 0 0.3502E-03 300.00 end
```

```
u-236     1 0 0.1428E-03 300.00 end
u-238     1 0 0.2146E-01 300.00 end
np-237    1 0 0.1383E-04 300.00 end
pu-238    1 0 0.4534E-05 300.00 end
pu-239    1 0 0.1373E-03 300.00 end
pu-240    1 0 0.5351E-04 300.00 end
pu-241    1 0 0.3208E-04 300.00 end
pu-242    1 0 0.1127E-04 300.00 end
am-241    1 0 0.9976E-06 300.00 end
am-242    1 0 0.2071E-07 300.00 end
am-243    1 0 0.2359E-05 300.00 end
cm-242    1 0 0.3017E-06 300.00 end
cm-244    1 0 0.6846E-06 300.00 end
i-135     1 0 0.2543E-07 300.00 end
'Zirc
cr        2 0 7.5891E-5   300.0 end
fe        2 0 1.4838E-4   300.0 end
zr        2 0 4.2982E-2   300.0 end
'Water w/ 2000 ppm boron
h2o       3 0.99 300.0 end
'b-10     3 0 2.2061E-5   300.0 end
'SS structural material
ss304     4 0.99 300.0 end
'Boral (model as b4c-al using areal density of b-10 @ -- g/cm^2 and 0.18 atom percent b-10 in nat. b)
'Excluded Proprietary Information
end comp
'squarepitch card excluded - Proprietary Information
more data
dab=999
end more
read param
gen=103 npg=3000 xs1=yes pki=yes gas=yes flx=yes fdn=yes far=yes nb8=999
end param
read geom
'geom cards excluded - Proprietary Information
end geom
read array
ara=1 nux=15 nuy=15 nuz=1 fill
```

```
   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   1  1  2  1  1  2  1  1  1  2  1  1  2  1  1
   1  1  1  1  1  1  1  2  1  1  1  1  1  1  1
   1  1  1  1  2  1  1  1  1  1  2  1  1  1  1
   1  1  2  1  1  1  1  1  1  1  1  1  2  1  1
   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   1  1  1  2  1  1  1  2  1  1  1  2  1  1  1
   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   1  1  2  1  1  1  1  1  1  1  1  1  2  1  1
   1  1  1  1  2  1  1  1  1  1  2  1  1  1  1
   1  1  1  1  1  1  1  2  1  1  1  1  1  1  1
   1  1  2  1  1  2  1  1  1  2  1  1  2  1  1
   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
end fill
end array
read bounds all=mirror end bounds
read mixt sct=2 eps=1.e-01 end mixt
read plot
scr=yes
ttl='w15x15 in High Density Rack'
xul=-11.5 yul= 11.5 zul=0.0
xlr= 11.5 ylr=-11.5 zlr=0.0
uax=1 vdn=-1 nax=750
end plot
end data
end
```

Skipped Cells	PWR	BWR
2	1.03533	1.02628
4	1.01192	1.01503
6	1.00363	1.01218
8	0.99786	1.01059

Table 1 Eigenvalue (using infinite multiplication factor) reduction from skipping cells between high reactivity assemblies.

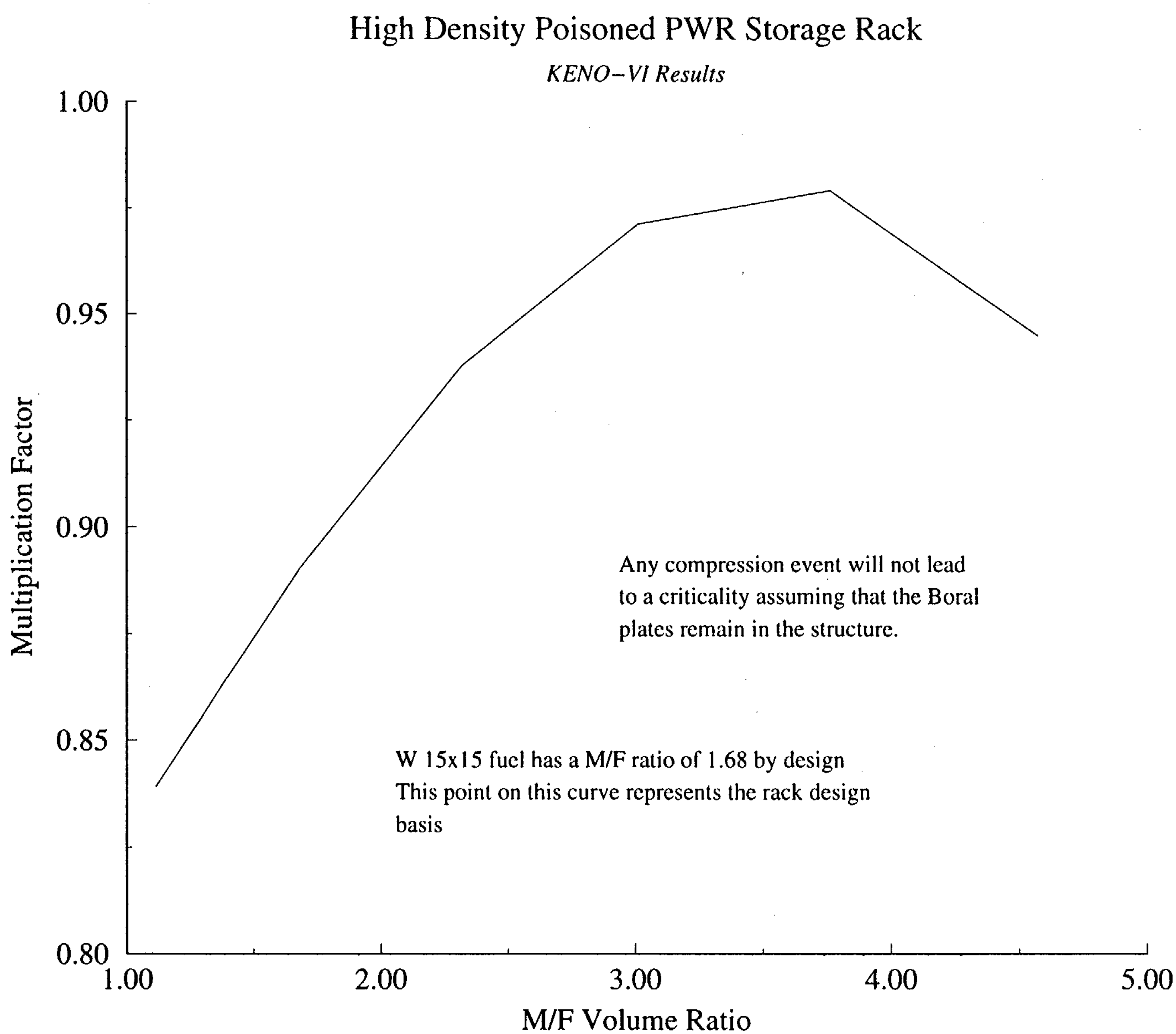

Figure 1 PWR High Density Storage Rack Eigenvalue following Compressive/Expansion Events

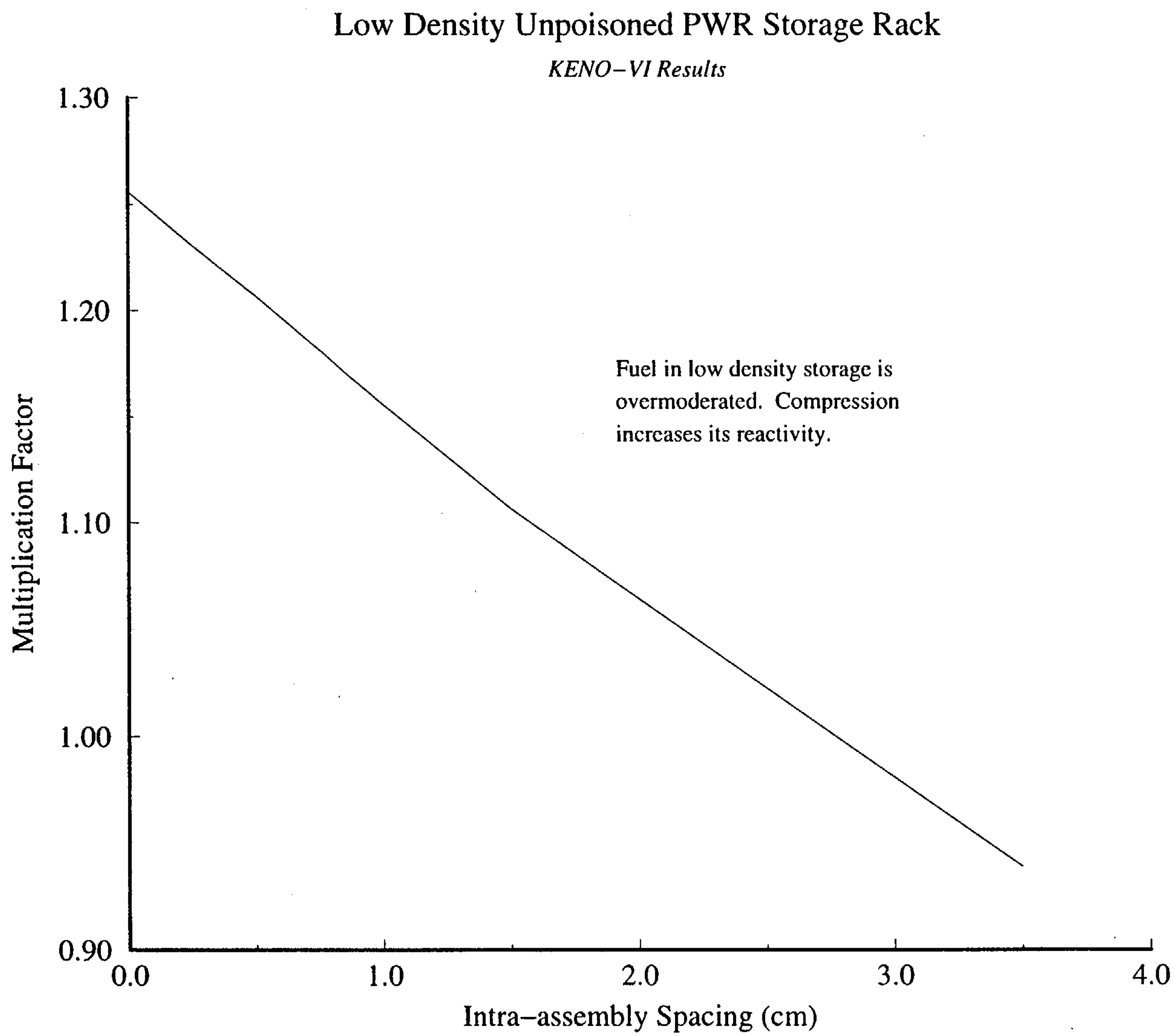

Figure 2 PWR Low Density Storage Rack Eigenvalue following Compressive/Expansion Events

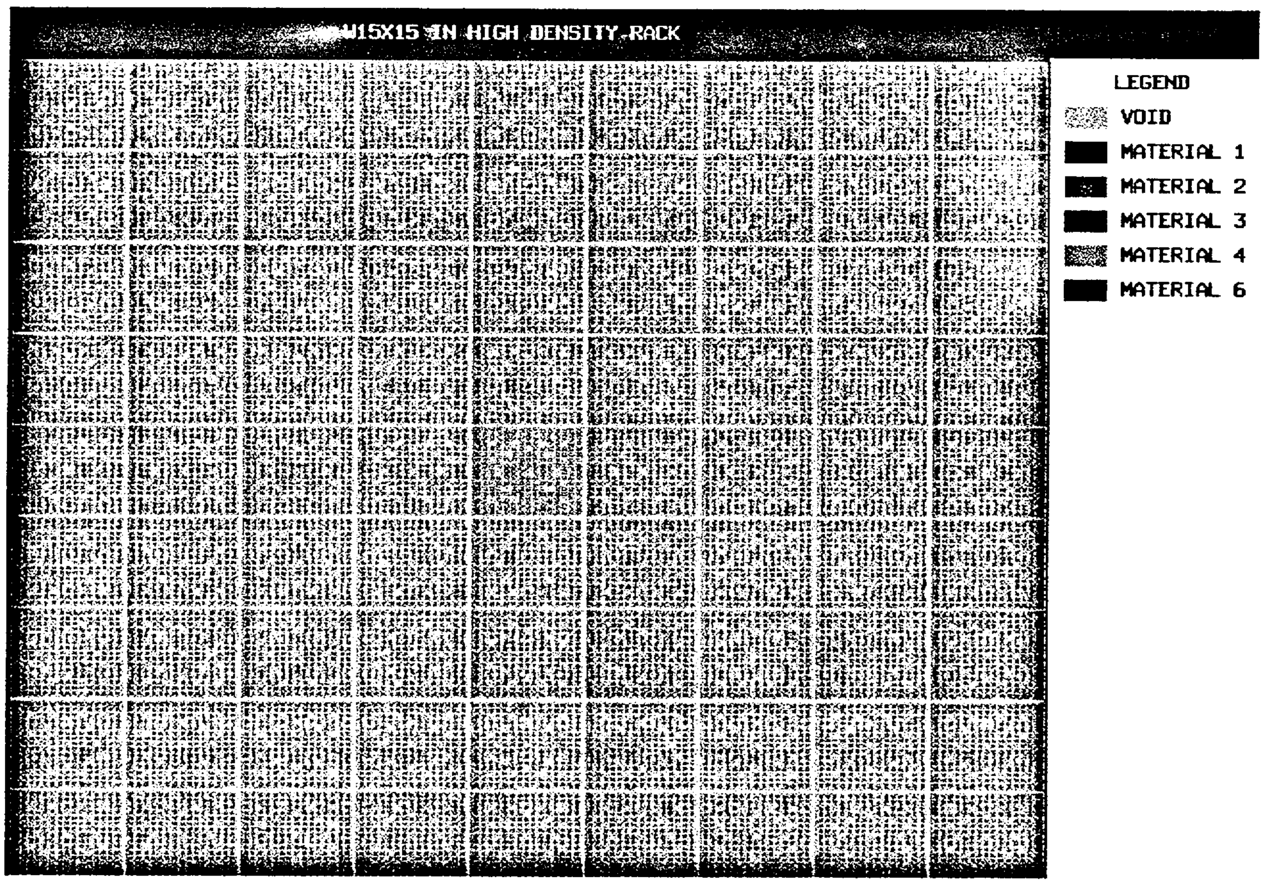

Figure 3 Sample Geometry Assuming 4 Assembly Spacing Between Most Reactive Assembly

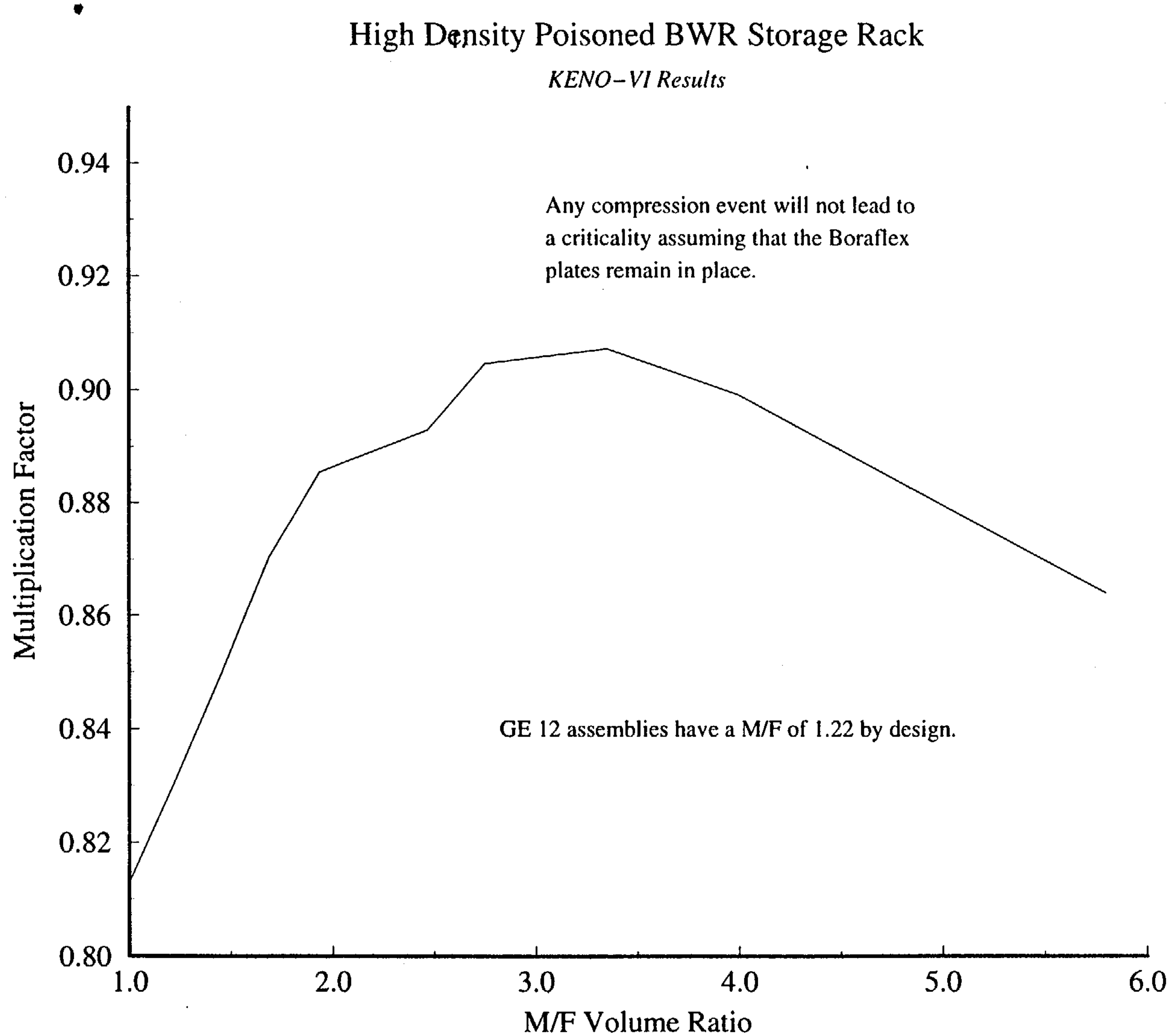

Figure 4 BWR High Density Storage Rack Eigenvalue following Compressive/Expansion Events

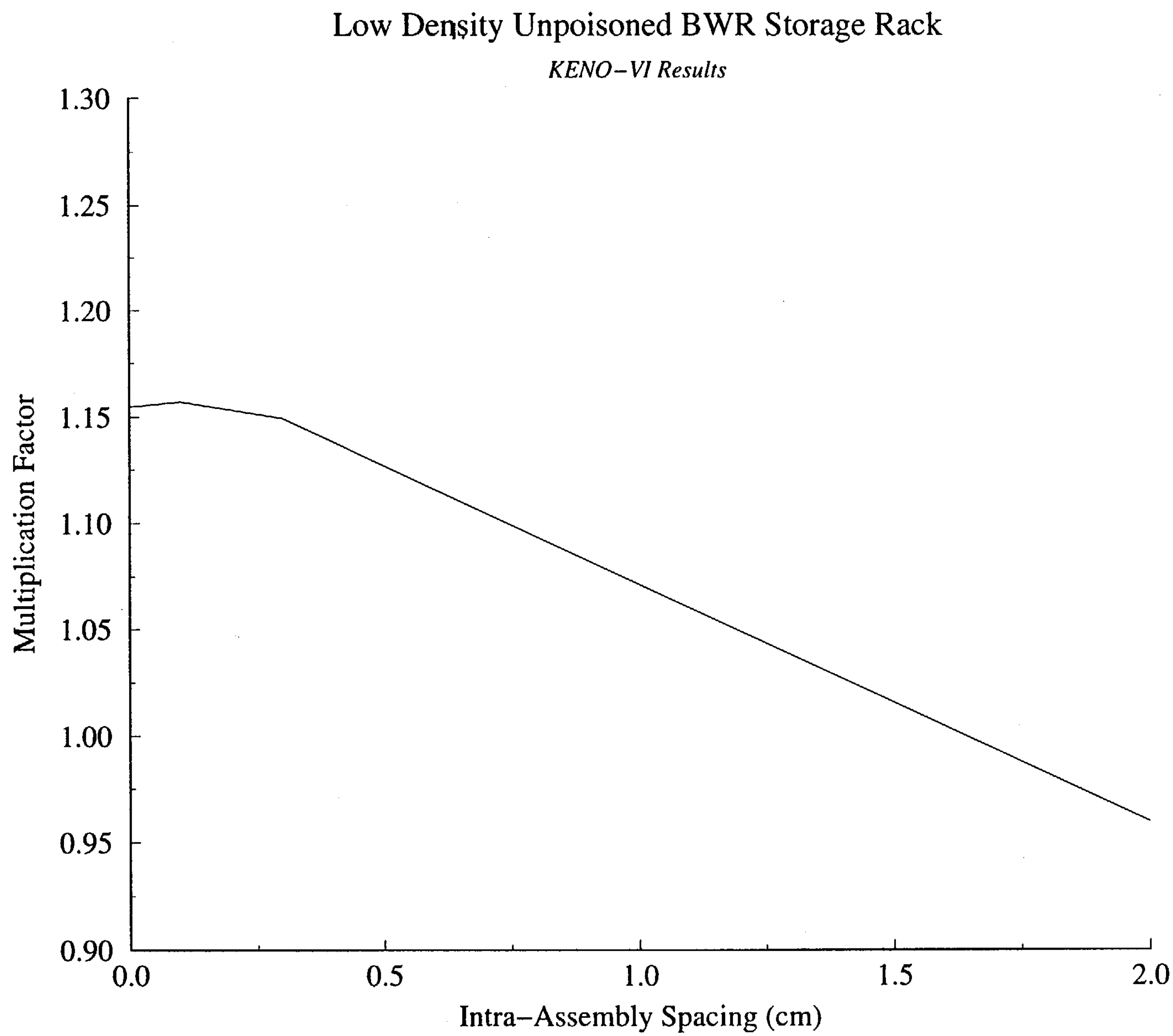

Figure 5 BWR Low Density Storage Rack Eigenvalue following Compressive/Expansion Events