SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
SECOND RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

ATTACHMENT 4
CRITICALITY SAFETY ANALYSES OF BWR FUEL WITHOUT CREDIT FOR BORAFLEX IN THE RACKS AT THE HARRIS NUCLEAR POWER STATION

Holtec Report No. HI-2043321, Revision 6 (Non-Proprietary)
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FOR

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Summary of Revisions

Revision 0:

Original issue.

Revision 1:

Client Revised GE13 Fuel Assembly Design and Axial Segment Description.

Revision 2:

Incorporated client comments; expanded discussion on axial burnup distribution and selection of the reference assembly.

Revision 3:

Revised Appendix A to remove the Propriety Information note in the footer.

Revision 4:

Revised to increase the burnup requirements for enrichments of 1.5, 2.0 and 2.5 wt% to reduce the maximum k_{eff} below 0.99. The enrichment tolerance is changed to [redacted]. The polynomial burnup versus enrichment curve is recalculated as a linear fit. Figure 2 is replaced.

Revision 5

Revised to incorporate additional calculations to support NRC acceptance review questions.

Revision 6

Revised to incorporate new curves and tables for 4 and 7 year cooling times. Revision includes changes made to HI-2043306R6, R7 and R8.

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1.0 PURPOSE

The purpose of the present evaluation is to document the criticality safety of the BWR fuel storage racks in the spent fuel pools of the Harris Plant. The pool criticality analyses are performed under the very conservative assumption of the complete loss of Boraflex in the BWR storage rack. With the assumed loss of all Boraflex material, the temperature coefficient of reactivity is positive. Therefore the limiting calculations assumed a temperature of 150 °F, which is an administrative limit for the spent fuel pool. Higher temperatures are considered accident conditions for which the soluble boron normally present in the pool water would assure the reactivity is maintained below the regulatory limit of a k_{eff} of less than 0.95. Under normal storage conditions, partial credit is taken for the soluble boron in the pool water, credit for fuel burnup is taken, and credit for spent fuel cooling time is also taken. The criticality analyses use the MCNP4a code, a Monte Carlo code developed by the Los Alamos National Laboratory, with explicit modeling of actinide and fission product nuclide concentrations. CASMO4 was used for calculation of manufacturing tolerances, and to determine the burnup dependent nuclide inventories used in the MCNP4a calculations at the various burnups.

Benchmark calculations, presented in Appendix A, indicate a bias of 0.0012 with an uncertainty of ± 0.0090 for MCNP4a, evaluated with a 95% probability at the 95% confidence level(1). The calculations for this analysis utilize the same computer platform and cross-section libraries used for the benchmark calculations discussed in Appendix A.

Benchmark calculations, presented in Reference 6, indicate a negative bias and bias uncertainty of ± 0.0025 for CASMO-4 evaluated with a 95% probability at the 95% confidence level(1). Since CASMO-4 is used to determine reactivity differences, the bias does not need to be applied to the results of the calculations. However, the bias uncertainty is included with the other uncertainties when determining the maximum k_{eff} values.

As described in the USNRC guidelines, parametric evaluations were performed independently for each of the manufacturing tolerances and the associated reactivity uncertainties were combined statistically. All calculations were made for an explicit modeling of the fuel and storage cell to define the limiting enrichment-burnup combinations for spent fuel that assures the safe storage of spent fuel in the BWR racks.

The criticality safety criteria used in the analysis was (1) the racks remain subcritical without any credit for the soluble boron present and (2) partial credit is taken for the soluble boron to assure the reactivity remains below 0.95 under normal and accident conditions.

The maximum k_{eff} values were determined assuming an infinite radial array of storage cells with a finite axial length, water reflected. For each initial enrichment, a minimum burnup was determined that assures the maximum k_{eff}, including calculational and manufacturing uncertainties, remains sub-critical under the assumed absence of all soluble boron. A conservative axial burnup distribution (Figure 1) was used in the calculations. Figures 2 and 2a summarize the results of these analyses, showing the minimum acceptable burnup for fuel of various initial maximum planar average enrichments at cooling times of 4 and 7 years, respectively. The limiting points in Figures 2 and 2a may be fitted by the following polynomial functions of the initial maximum planar average enrichment, E.

4 years: Burnup Limit = $0.5406 * E^3 - 5.3804 * E^2 + 33.389 * E - 38.167$

7 years: Burnup Limit = $0.3850 * E^3 - 3.8506 * E^2 + 28.063 * E - 32.905$

The polynomials were selected so all points are bounded. Table 5 shows the calculated minimum burnups and the values determined using the polynomials for both cooling times.

The soluble boron concentration required to maintain k_{eff} below 0.95, including all manufacturing and calculation tolerances, for storage of fuel in the pool was determined to be 300 ppm under normal conditions. An accident scenario, where a maximum reactivity fuel assembly is accidentally loaded into an otherwise filled rack was also evaluated. For this accident case, 325 ppm soluble boron would be required to maintain k_{eff} below 0.95. This minimum soluble boron concentration of 325 ppm is well below the 2000 ppm soluble boron administrative limit for the pool water. This provides a large safety margin in reactivity.

Based on the analyses presented herein, it is concluded that the BWR Boraflex spent fuel storage racks can safely accommodate fuel with initial maximum planar average enrichments up to 4.6%, with assurance that the maximum reactivity, including calculational and manufacturing uncertainties, will be less than 0.95, with 95% probability at the 95% confidence level, provided only that (1) the fuel conforms to the enrichment-burnup limits for the spent fuel as depicted in Figures 2 and 2a, and (2) that a minimum of 325 ppm soluble boron is maintained. The limiting burnups shown in Figures 2 and 2a, and in Table 5 are the assembly average burnups and, in their application, must be adjusted for the plant's uncertainty in determining the actual burnups of the spent fuel assemblies.

2.0 ANALYSIS CRITERIA AND ASSUMPTIONS

To assure the true reactivity will always be less than the calculated reactivity, the following conservative analysis criteria or assumptions were used.

- Criticality safety analyses were based upon an infinite radial array of cells; i.e. no credit was taken for radial neutron leakage.

- Minor structural materials were neglected; i.e. spacer grids were conservatively assumed to be replaced by water.

- Because the temperature coefficient of reactivity is positive in the absence of Boraflex, the analyses assumed the administrative limit temperature of 150 °F. Higher temperatures would be an accident condition for which soluble boron credit is permitted.

- The axial burnup distribution calculations were performed assuming an axial distribution shown in Figure 1 (and Table 6).

3.0 ACCEPTANCE CRITERIA

The primary acceptance criterion under normal conditions is that the maximum k_{eff} shall be less than critical, including calculational uncertainties and effects of mechanical tolerances under the postulated absence of all soluble boron. Partial credit is taken for the soluble boron in the pool water to assure that the maximum k_{eff} shall be less than 0.95, including calculational uncertainties and effects of mechanical tolerances, under normal and accident conditions.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.
- Code of Federal Regulation 10CFR50.68, Criticality Accident Requirements.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage.
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications.
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (draft), December 1981.
- ANSI-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum L. Kopp to Timothy Collins, August 19, 1998.

4.0 DESIGN AND INPUT DATA

4.1 FUEL ASSEMBLY DESIGN

The design basis fuel assembly is the GE13 assembly, a 9x9 array of UO_2 fuel rods and 2 large water rods. Table 2 provides the pertinent design details for the GE13 fuel assembly as well as other fuel designs stored in the Harris spent fuel pools. Tables 7 and 7a provide comparisons of the reactivities of the fuel assembly designs present in the spent fuel pool for enrichments between 2.0 wt% and 4.6 wt% ^{235}U for burnups above and below the minimum burnup requirement. Table 7 shows that for a cooling time of 4 years the GE 3, GE4 and GE 7 assemblies have higher reactivities than the GE13 assembly. However, these fuel assembly designs were used in earlier operation of the Brunswick reactor and these assemblies have cooling times in excess of 26, 23 and 12 years for the GE3, GE4 and GE7 assemblies, respectively. Table 7 shows that when cooling time is also considered for these fuel assembly designs, the GE13 assembly with 4 years cooling time has the highest reactivity. Table 7a shows a similar comparison for 7 years cooling time for the GE13 assembly. Therefore, the GE13 assembly is used in all further calculations described in subsequent sections.

The axial dimensions of the GE13 fuel assembly are:

GE-13 Bundle Axial Description		
Height (from bottom of bundle, in inches)	Lattice Description	Number of fuel rods
0.0 to 6.0	Natural Uranium Blanket	74 fuel rods
6.0 to 108.0	Full Lattice	74 fuel rods
108.0 to 138.0	Part-Length Lattice	66 fuel rods
138.0 to 146.0	Natural Uranium Blanket	66 fuel rods

4.2 CORE OPERATION AND CONTROL RODS

Core operating parameters are necessary for fuel depletion calculations performed with CASMO-4. The core parameters used for the depletion calculations are presented in Table 2a.

Temperature and void fraction values are taken as the upper bound (most conservative) of the core operating parameters of Brunswick. The neutron spectrum is hardened by each of these parameters, leading to a greater production of plutonium during depletion, which results in conservative reactivity values.

Additionally, the Brunswick reactor uses control rod blades for reactor and power control during operation. The geometrical and material properties of the control rods are provided in Table 2b. The control rod operating strategy at Brunswick for GE 13 fuel did not allow fresh fuel to be placed in a core location that would have planned control rod insertion. Therefore, control rod insertion is limited to once and twice burned fuel assemblies. Typical fuel is controlled for 3 GWD/MTU intervals then uncontrolled for 3 GWD/MTU intervals. Therefore, to conservatively bound any control rod insertion, fuel assemblies are modeled with an initial interval of 12 GWD/MTU (i.e., first cycle) of uncontrolled operation followed by intervals of 3 GWD/MTU controlled and uncontrolled operation. This is conservative as fuel is not actually controlled for 3 GWD/MTU as the flux suppression of the control rod blade significantly decreases exposure accumulation.

4.3 STORAGE RACK DESIGN

The storage cells are composed of stainless steel boxes, joined at the corners in an egg-crate structure. Initially the design included Boraflex as the absorber, although in the present analyses, the Boraflex is assumed to be lost. The storage cells are located on a lattice spacing of 6.25 inches (for conservatism, a lattice spacing of 6.22 inches was used in the analyses). The box wall thickness is 0.075 inches, and the box inside dimension is 6.05 inches. The wrapper wall thickness is 0.035 inches. The Boraflex panels were assumed to be completely replaced by water. A cross-section of the storage cell is shown in Figure 3.

5.0 METHODOLOGY

5.1 GENERAL DESCRIPTION

The primary criticality analyses (at 95% probability, 95% confidence level)[1] were performed with the three-dimensional MCNP4a code[2]. Benchmark calculations (Appendix A) have determined a calculational bias of 0.0012±0.0090 Δk for MCNP4a calculations.

CASMO4, a two-dimensional deterministic code[3] using transmission probabilities, was used to evaluate the small (differential) reactivity effects of manufacturing tolerances and to determine nuclide concentrations developed in the depletion calculations.

In the geometric model used in the calculations, each fuel rod and each fuel assembly were explicitly described. Reflecting boundary conditions effectively defined an infinite radial array of storage cells. In the axial direction, a 30-cm water reflector was used to conservatively describe axial neutron leakage. Each stainless steel box and water gap were also described in the calculational model. The fuel cladding material was zirconium.

MCNP4a Monte Carlo calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To assure convergence and to minimize the statistical uncertainty of the calculated reactivities, a minimum of 4 million neutron histories was accumulated in each calculation, generally resulting in a statistical uncertainty of about ±0.0003 Δk.

5.2 AXIAL BURNUP DISTRIBUTION

Usually BWR storage rack analyses do not credit soluble boron in the pool water and use low burnup fuel with credit for gadolinia. However, in the Harris spent fuel pool, soluble boron is present and higher burnup fuel is necessarily credited. With high burnup fuel, the axial distribution in burnup becomes important and must be considered in any assessment of reactivity.

Progress Energy provided 16 axial burnup distributions for assemblies of approximately 4.03 wt% enrichment with burnups between 30.2 GWD/MTU and 45.9 GWD/MTU. The minimum required burnup for assemblies with 4.0 wt% enrichment as shown in Table 5 is 44.0 GWD/MTU. Therefore, four axial burnup distributions with an assembly average burnup near or below the minimum required burnup specified in Table 5 were chosen and averaged to determine an appropriate axial burnup distribution.

The axial burnup distribution is conservative because the profiles selected for averaging are from assemblies with burnups near or below the required minimum burnup. Assemblies with much lower burnups, which would have profiles that might produce higher reactivities, would not be able to be stored in the racks because they would not meet the burnup requirement. Assemblies with higher burnups have profiles that are more cosine shaped, which would further reduce the reactivity. This is shown in the following table where the resultant reactivity of an axial burnup distribution from an assembly with an enrichment of approximately 4.0 wt% with a burnup of 45.9 GWD/MTU is compared to the resultant reactivity from an assembly with the axial burnup distribution in Table 6. These calculations were performed at a burnup of 45 GWD/MTU with an enrichment of 4.0 wt% ^{235}U.

Profile	Table 6	YJM160
Assembly Burnup [GWD/MTU]	37.5[1]	45.88
Calculated k_{eff}	0.9554[2]	0.9186

Therefore, based on the level of conservatism inherent in choosing the axial burnup distribution in the manner described above, it is not necessary to confirm that the axial burnup distributions of individual assemblies are bounded by the assumed axial burnup distribution. Additionally, other areas of conservatism such as the assumed reactor records burnup uncertainty, longer cooling times than the assumed 4 or 7 years, and the presence of soluble boron further insure that the BWR racks meet the acceptance criteria stated in Section 3.0.

[1] The assembly burnup for the burnup profile in Table 5 is an average of the assembly average burnups from the four assemblies selected to determine the axial burnup profile in Table 5.
[2] This is the calculated k_{eff} from Table 1 for 4.0 wt% at 45 GWD/MTU

In the present analyses, the final axial burnup distribution assumed is illustrated in Figure 1, as determined by 25 segment calculations. Four different axial burnup distributions for GE13 fuel were selected. The normalized axial burnup distributions are very nearly the same. An average of the four axial burnup distributions for GE13 fuel was developed and used in the present analyses as illustrated in Figure 1 and listed in Table 6.

These calculations use axial blankets of natural uranium oxide and include the effect of part-length fuel rods. The lower fully rodded zone (up to 108 inches above the bottom of the fuel) contains 74 fuel rods and the partially-rodded zone contains 66 fuel rods. At the top of the assembly, the blanket is 8 inches in length and the bottom blanket is 6 inches long. The top axial blanket is extended further 4 inches (to make the total fuel height 150 inches) so that the axial burnup distribution and the fuel height modeled are consistent.

Separate CASMO4 depletion calculations were made for each of the 25 axial segments and the isotopic composition of each of the 25 segments were transferred to a 3-dimensional MCNP4a case, thereby inherently incorporating the effect of the axial burnup distribution.

In some cases of lower average burnups, a uniform axial burnup distribution can result in a higher reactivity than the distributed burnup case. Therefore, all calculations were also performed with an axially flat profile equal to the assembly average burnup. The profile (distributed or flat) which produced the highest reactivity was used to determine the maximum k_{eff} in the storage racks.

6.0 ANALYSIS RESULTS

6.1 EVALUATION OF TOLERANCE UNCERTAINTIES

CASMO4 calculations were made to determine the uncertainties in reactivity associated with manufacturing tolerances. Results of these calculations are shown in Tables 1, 1a and 3. The reactivity effects were separately evaluated, in a sensitivity study for each independent tolerance, and the results combined statistically, using the root mean square methodology. Tolerances considered include the following:

- Tolerance in pitch – minimum pitch (6.22") used which is less than the nominal pitch (6.25")
- Tolerance in Steel Box I.D.
- Tolerance in Stainless Steel Box Wall Thickness
- Tolerance in Uranium Enrichment
- Tolerance in UO_2 Density

In addition to these mechanical tolerances, uncertainties due to tolerance in benchmarking calculations, statistical variation in MCNP4a calculations and the estimated uncertainty in depletion calculations (section 6.1.1 below) are included (See Tables 1 and 1a). Also included is the incremental reactivity between 20 °C (MCNP4a calculation) and the reference temperature of 150 °F. The effect of soluble boron on the manufacturing tolerances is not considered in the BWR racks since the soluble boron requirement is low (less than 325ppm) compared with the plant Technical Specification.

6.1.1 UNCERTAINTY IN DEPLETION CALCULATIONS

The uncertainty in depletion calculations were taken as 5% of the reactivity decrement from beginning-of-life to the burnup of concern[(4)]. These uncertainties depend on burnup and are listed in Table 1 and 1a.

6.2 ABNORMAL AND ACCIDENT CONDITIONS

A brief summary of the calculated reactivity effects of the accident conditions is given in Table 4.

6.2.1 ECCENTRIC LOCATION OF FUEL ASSEMBLIES

The fuel assemblies are normally stored in the center of the storage cells. Calculations were made with the fuel assemblies assumed to be in the corner of the storage rack cell (eccentric positioning of a four-assembly cluster at closest approach). As shown in Figure 3, reflective boundary conditions are used on the periphery of the single cell model. This creates an infinite array of storage cells. By moving the assembly in the single storage cell model to the corner of the storage cell, an infinite array of storage cells is created with each cluster of four assemblies being placed closest to a common corner. Eccentric positioning of spent fuel assemblies resulted in a slightly lower reactivity in these racks.

6.2.2 TEMPERATURE AND VOID EFFECTS

Temperature effects were also evaluated using CASMO4. These results presented in Table 8 show that the temperature coefficient of reactivity is positive and the reactivity at 150 °F (maximum expected spent fuel pool water temperature) is used to derive a bias. Any residual Boraflex that might remain would reduce the temperature penalty. At the submerged depth of the storage pool, the maximum temperature at boiling is 120 °C, and the void coefficient of reactivity is negative.

The reference MCNP4a calculations were performed at a water density corresponding to a temperature of 20 °C. The reactivity increment between that at the maximum expected water temperature and at 20 °C is taken into account as an additive term (bias). This bias, at each enrichment considered, is listed in Table 1 and 1a.

6.2.3 MIS-LOADED FUEL ASSEMBLY ACCIDENT

The potential effects of a fuel mis-loading accident condition were also considered in this study. To evaluate the consequence of the fuel mis-loading accident, the misloaded fuel assembly was assumed to be the most reactive assembly possible – a spent fuel assembly with a k_∞ of 1.33[3] in the Standard Cold Core Geometry.[*] Fuel isotopes (not including any gadolinia) were extracted from the CASMO4 calculation and used in a 3-dimensional MCNP4a calculational model. All other assemblies in the calculation used the CASMO4 nuclide inventory for the normal fuel burnup.

For the most serious postulated mis-loading accident scenario, calculations were performed to determine the soluble boron concentration required to maintain k_{eff} below 0.95 in the pool under the postulated accident scenario. Results of this calculation showed that 325 ppm soluble boron would be adequate to assure a k_{eff} less than 0.95.

7.0 CONCLUSIONS

Fuel assemblies with spent fuel having at least the burnup-enrichment combination as depicted in Figures 2 and 3, and Table 5 may be safely accommodated in the storage racks, with no other constraints on their placement in the pool.

[3] The SCCG k_{inf} of 1.33 is based on the maximum SCCG k_{inf} provided in Table 2 (1.32) plus an additional 0.01 Δk adder to account for differences in the calculation of the SCCG k_{inf} between Holtec and the fuel vendor.

[*] The k_∞ (SCCG) is defined as the reactivity of an infinite array of assemblies at 20 °C on a 6-inch lattice pitch without void or any control element.

8.0 REFERENCES

[1] M.G. Natrella, Experimental Statistics, National Bureau of Standards, Handbook 91, August 1963.

[2] J. F. Briesmeister, editor, *MCNP - A General Monte Carlo N-Particle Transport Code,* LA-12625-M, Los Alamos National Laboratory, November 1993.

[3] A. Ahlin, M. Edenius, H. Haggblom, "CASMO- A Fuel Assembly Burnup Program," AE-RF-76-4158, Studsvik report (proprietary).

A. Ahlin and M. Edenius, "CASMO- A Fast Transport Theory Depletion Code for LWR Analysis," ANS Transactions, Vol. 26, p. 604, 1977.

D. Knott, "CASMO4 Benchmark Against Critical Experiments", Studsvik Report SOA-94/13 (Proprietary).

M. Edenius et al., "CASMO4, A Fuel Burnup Program, Users Manual" Studsvik Report SOA/95/1.

[4] L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum, L. Kopp to Timothy Collins, August 19, 1998.

[5] "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit," NUREG/CR-6760, ORNL/TM-2000/231, March 2002.

[6] HI-2094370R0, "CASMO-4 Benchmark for Spent Fuel Pool Criticality Analysis."

Table 1 : Maximum Reactivity at Various Initial Enrichment – Minimum Burnup Combinations (4 years cooling)

Initial Enrichment, wt% U-235	1.50%	2.00%	2.50%	3.00%	3.50%	4.00%	4.60%
Minimum Burnup,MWD/KgU	1.5	11.1	20.1	27.6	35.5	43.9	53.8
MCNP Bias	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012
Temperature Correction to 150°F	0.0044	0.0061	0.0067	0.0069	0.0071	0.0071	0.0071
Calculated k_{eff}	0.9604	0.9636	0.9634	0.9630	0.9615	0.9609	0.9600
Uncertainties							
MCNP Bias Uncertainty	0.0090	0.0090	0.0090	0.0090	0.0090	0.0090	0.0090
CASMO Bias Uncertainty	0.0025	0.0025	0.0025	0.0025	0.0025	0.0025	0.0025
Calculational	0.0006	0.0006	0.0006	0.0006	0.0006	0.0008	0.0006
Eccentricity	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Manufact. Tolerances	0.0219	0.0157	0.0129	0.0118	0.0108	0.0099	0.0094
Depletion Uncertainty	0.0006	0.0054	0.0088	0.0115	0.0137	0.0154	0.0171
Total Uncertainty	0.0238	0.0191	0.0183	0.0190	0.0198	0.0206	0.0216
Maximum k_{eff}	0.9898	0.9900	0.9895	0.9900	0.9896	0.9897	0.9900

Table 1a : Maximum Reactivity at Various Initial Enrichment – Minimum Burnup Combinations (7 years cooling)

Initial Enrichment, wt% U-235	1.50%	2.00%	2.50%	3.00%	3.50%	4.00%	4.60%
Minimum Burnup,MWD/KgU	1.4	10.5	19.2	26.2	34.2	42.3	51.7
MCNP Bias	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012
Temperature Correction to 150°F	0.0044	0.0060	0.0071	0.0067	0.0074	0.0070	0.0070
Calculated k_{eff}	0.9605	0.9633	0.9633	0.9629	0.9615	0.9608	0.9601
Uncertainties							
MCNP Bias Uncertainty	0.0090	0.0090	0.0090	0.0090	0.0090	0.0090	0.0090
CASMO Bias Uncertainty	0.0025	0.0025	0.0025	0.0025	0.0025	0.0025	0.0025
Calculational	0.0006	0.0006	0.0006	0.0006	0.0006	0.0006	0.0006
Eccentricity	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Manufact. Tolerances	0.0219	0.0159	0.0126	0.0118	0.0106	0.0100	0.0094
Depletion Uncertainty	0.0006	0.0054	0.0089	0.0115	0.0137	0.0154	0.0171
Total Uncertainty	0.0239	0.0192	0.0180	0.0190	0.0197	0.0206	0.0216
Maximum k_{eff}	0.9899	0.9897	0.9896	0.9898	0.9898	0.9897	0.9899

Table 2: BWR Fuel Characteristics

Fuel Assembly	GE 3	GE 4	GE 7	GE 8	GE 9	GE 10	GE 13
NOTE: All dimensions in inches							
Clad O.D.	0.563	0.493	0.483	0.483	0.483	0.483	0.440
Clad I.D.	0.489	0.425	0.419	0.419	0.419	0.419	0.384
Clad Material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
Pellet Diameter	0.477	0.416	0.410	0.411	0.411	0.411	0.376
Stack Density	10.31	10.40	10.54	10.58	10.54	10.54	10.54
Maximum Enrichment	4.6	4.6	4.6	4.6	4.6	4.6	4.6
SCCG k_{inf}	≤1.32	≤1.32	≤1.32	≤1.32	≤1.32	≤1.32	≤1.32
Active Fuel Length	144	146	150	150	150	150	146
Axial Length of Partial Rods, in	—	—	—	—	—	—	108
Fuel Rod Array	7x7	8x8	8x8	8x8	8x8	8x8	9x9
Number Fuel Rods	49	63	62	60	60	60	74/66
Fuel Rod Pitch	0.738	0.640	0.640	0.640	0.640	0.640	0.566
Number of Water Rods	0	1	2	4	1	1	2
Water Rod O.D.	NA	0.493	0.591	0.591/ 0.483	1.34	1.34	0.980
Water Rod I.D.	NA	0.425	0.531	0.531/ 0.431	1.26	1.26	0.920
Channel I.D.	5.278	5.278	5.278	5.278	5.278	5.278	5.278
Channel Thickness	0.080	0.080	0.080	0.080	0.080	0.070	0.070

Notes:
1. The GE 13 assembly has 8 part length rods, 108 inches in height.
2. The GE 5 and GE 6 are identical to the GE 7 for the fuel parameters listed.
3. This data was provided by Progress Energy. The enrichment is the maximum planar average enrichment.

Table 2a: Brunswick Core Operating Parameters for Depletion

Parameter	Value
Reactor Specific Power, MW/MTU	30
Maximum Core Plane Average Fuel Temperature, °F	1260
Core Average Moderator Temperature at the Top of the Active Region, °F	560
In-Core Assembly Pitch, Inches	6.00
Maximum Void Fraction	0.77

Table 2b: BWR Control Rod Characteristics

Parameter	Value
Central Support Tie Rod Span, inches	1.56
Wing Tip to Wing Tip Blade Span, inches	9.81
Blade Wing Thickness, inches	0.312
Blade Sheath Thickness, inches	0.056
Poison Tube Wall Thickness, inches	0.025
Poison Tube OD, inches	0.188
Absorber Zone Axial Length, inches	143
Number of B_4C Tubes (Per Wing)	21
B_4C Density, g/cc	1.76

Table 3: Tolerances and Their Reactivity Effects

Description	Tolerance	Reactivity Effect, Δk
1) Tolerance in Storage Box ID	[redacted]	See Table 1
2) Tolerance in Stainless Steel Wall Thickness	[redacted]	See Table 1
3) Tolerance in Uranium Enrichment	[redacted]	See Table 1
4) Tolerance in UO_2 Density	± 0.200	See Table 1
5) Uncertainty due to Fuel Eccentricity	NA	Negative
6) Uncertainty in Depletion Calculations	5% of Reactivity Decrement	See Table 1

Table 4: Summary of Abnormal/ Accident Conditions

Condition	Consequence
Temperature Increase	150 °F used for normal storage condition. Higher temperatures are accident conditions with credit for soluble boron allowed.
Void (Boiling)	Negative void coefficient of reactivity
Assembly Drop on Top of Rack	Negligible
Seismic Movement	Negligible
Mis-Loaded Fuel Assembly	Requires 325 ppm soluble boron

Table 5: BWR Boraflex Racks Burnup Versus Enrichment Requirement

Initial Maximum Planar Average Enrichment, wt% U-235	4 years cooling		7 years cooling	
	Calculated Burnup Limit MWD/KgU	Polynomial Fit, MWD/KgU	Calculated Burnup Limit MWD/KgU	Polynomial Fit, MWD/KgU
1.5	1.5	1.6	1.4	1.8
2.0	11.1	11.4	10.5	10.9
2.5	20.1	20.1	19.2	19.2
3.0	27.6	28.2	26.2	27.0
3.5	35.5	36.0	34.2	34.7
4.0	43.9	43.9	42.3	42.4
4.6	53.8	54.2	51.7	52.2

Table 6: Average Segment Axial Burnup Distribution

Segment Number	Axial Height from Top of Active Fuel (cm)	Relative Burnup
1	0 – 15.24	0.1740
2	15.24 – 30.48	0.2490
3	30.48 – 45.72	0.5560
4	45.72 – 60.96	0.7160
5	60.96 – 76.20	0.8600
6	76.20 – 91.44	0.9610
7	91.44 – 106.68	1.0290
8	106.68 – 121.92	1.0370
9	121.92 – 137.16	1.0860
10	137.16 – 152.40	1.1260
11	152.40 – 167.64	1.1570
12	167.64 – 182.88	1.1820
13	182.88 – 198.12	1.2050
14	198.12 – 213.36	1.2260
15	213.36 – 228.60	1.2430
16	228.60 – 243.84	1.2590
17	243.84 – 259.08	1.2720
18	259.08 – 274.32	1.2840
19	274.32 – 289.56	1.2930
20	289.56 – 304.80	1.2970
21	304.80 – 320.04	1.2930
22	320.04 – 335.28	1.2490
23	335.28 – 350.52	1.1300
24	350.52 – 365.76	0.8680
25	365.76 – 381.00	0.2530

Table 7: Assembly Design Reactivities (4 years)

Burnup [GWD/MTU]	GE3 4 years	GE3 26 years	GE4 4 years	GE4 23 years	GE7 4 years	GE7 12 years	GE8 4 years	GE9 4 years	GE10 4 years	GE13 4 years
Enrichment = 2.0 wt% ^{235}U										
10	0.9891	0.9719	0.9832	0.9674	0.9811	0.9726	0.9787	0.9777	0.9642	0.9794
12.5	0.9660	0.9418	0.9601	0.9378	0.9574	0.9454	0.9546	0.9536	0.9398	0.9555
Enrichment = 3.0 wt% ^{235}U										
25	0.9714	0.9318	0.9656	0.9291	0.9642	0.9442	0.9629	0.9618	0.9462	0.9628
27.5	0.9515	0.9071	0.9456	0.9047	0.9435	0.9212	0.9418	0.9407	0.9249	0.9421
Enrichment = 4.0 wt% ^{235}U										
40	0.9445	0.8925	0.9388	0.8908	0.9377	0.9113	0.9372	0.9361	0.9193	0.9369
42.5	0.9276	0.8722	0.9219	0.8709	0.9202	0.8919	0.9191	0.9180	0.9010	0.9192
Enrichment = 4.6 wt% ^{235}U										
50	0.9220	0.8639	0.9164	0.8628	0.9151	0.8853	0.9146	0.9135	0.9070	0.9145
52.5	0.9068	0.8460	0.9013	0.8451	0.8993	0.8681	0.8983	0.8971	0.8903	0.8985

Table 7a: Assembly Design Reactivities (7 years)

Burnup [GWD/MTU]	GE3 27 years	GE4 24 years	GE7 13 years	GE8 11 years	GE9 11 years	GE10 9 years	GE13 7 years
Enrichment = 2.0 wt% ^{235}U							
10.0	0.9715	0.9669	0.9717	0.9711	0.9702	0.9587	**0.9759**
12.5	0.9413	0.9372	0.9442	0.9440	0.9429	0.9319	**0.9505**
Enrichment = 3.0 wt% ^{235}U							
25.0	0.9309	0.9280	0.9423	0.9451	0.9441	0.9331	**0.9544**
27.5	0.9061	0.9035	0.9190	0.9218	0.9207	0.9100	**0.9326**
Enrichment = 4.0 wt% ^{235}U							
40.0	0.8913	0.8894	0.9087	0.9134	0.9124	0.9017	**0.9257**
42.5	0.8710	0.8694	0.8892	0.8937	0.8926	0.8821	**0.9072**
Enrichment = 4.6 wt% ^{235}U							
50.0	0.8626	0.8612	0.8825	0.8879	0.8868	0.8870	**0.9018**
52.5	0.8446	0.8435	0.8651	0.8702	0.8690	0.8692	**0.8852**

Table 8: Reactivity Effect of Temperature and Void Content
4.0 wt%, 45.0 MWD/kgU

Temperature [°C]	Δk
0	Reference
4	+ 0.0007
26.9	+ 0.0049
65.6 (150 °F)	+ 0.0115
123	+ 0.0214
123, 10% Void	+ 0.0176

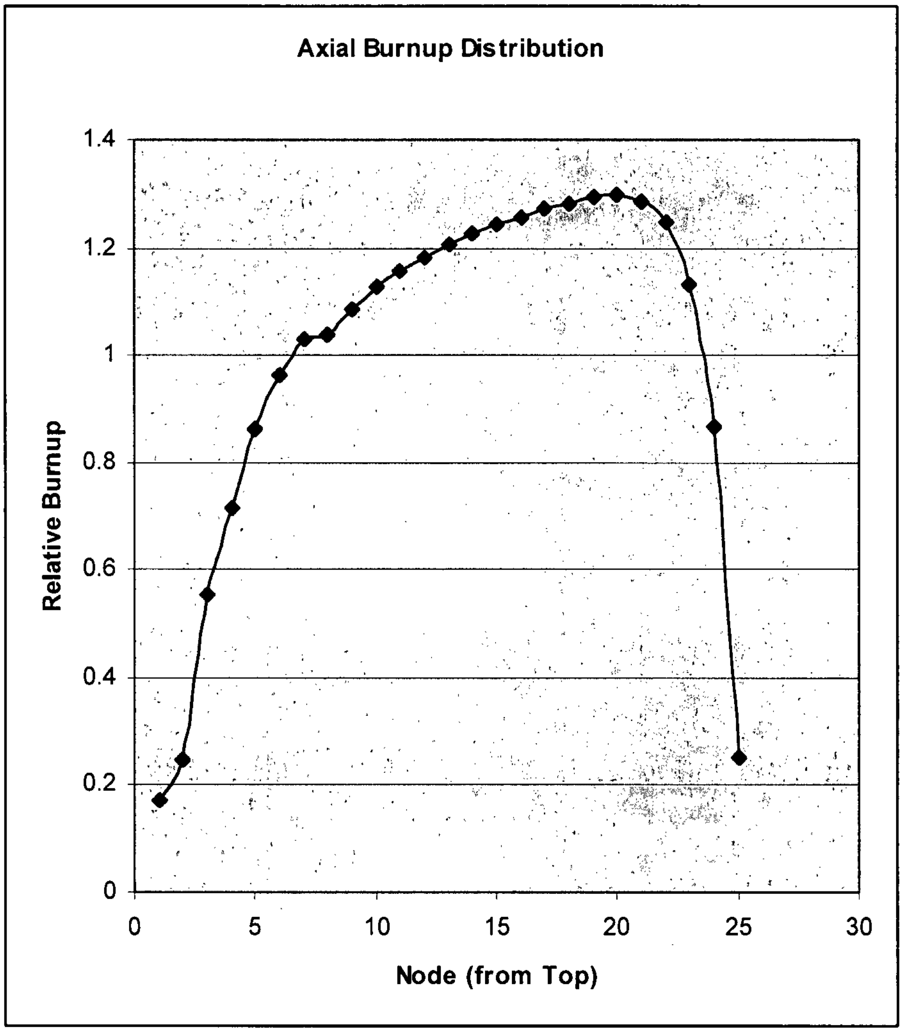

Figure 1: Axial Burnup Distribution

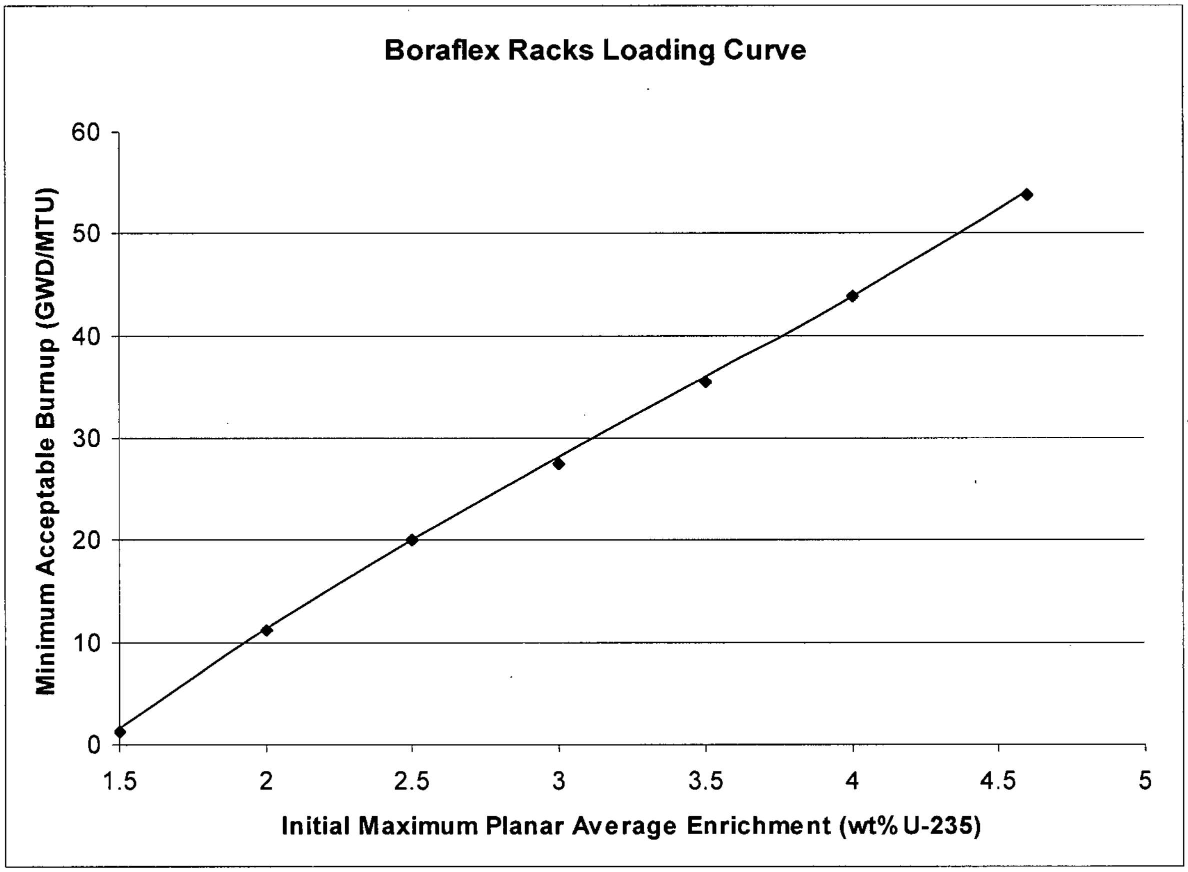

Fig. 2: Minimum Fuel Burnup for Acceptable Storage of Spent Fuel of Various Initial Planar Average Enrichments (4 years).

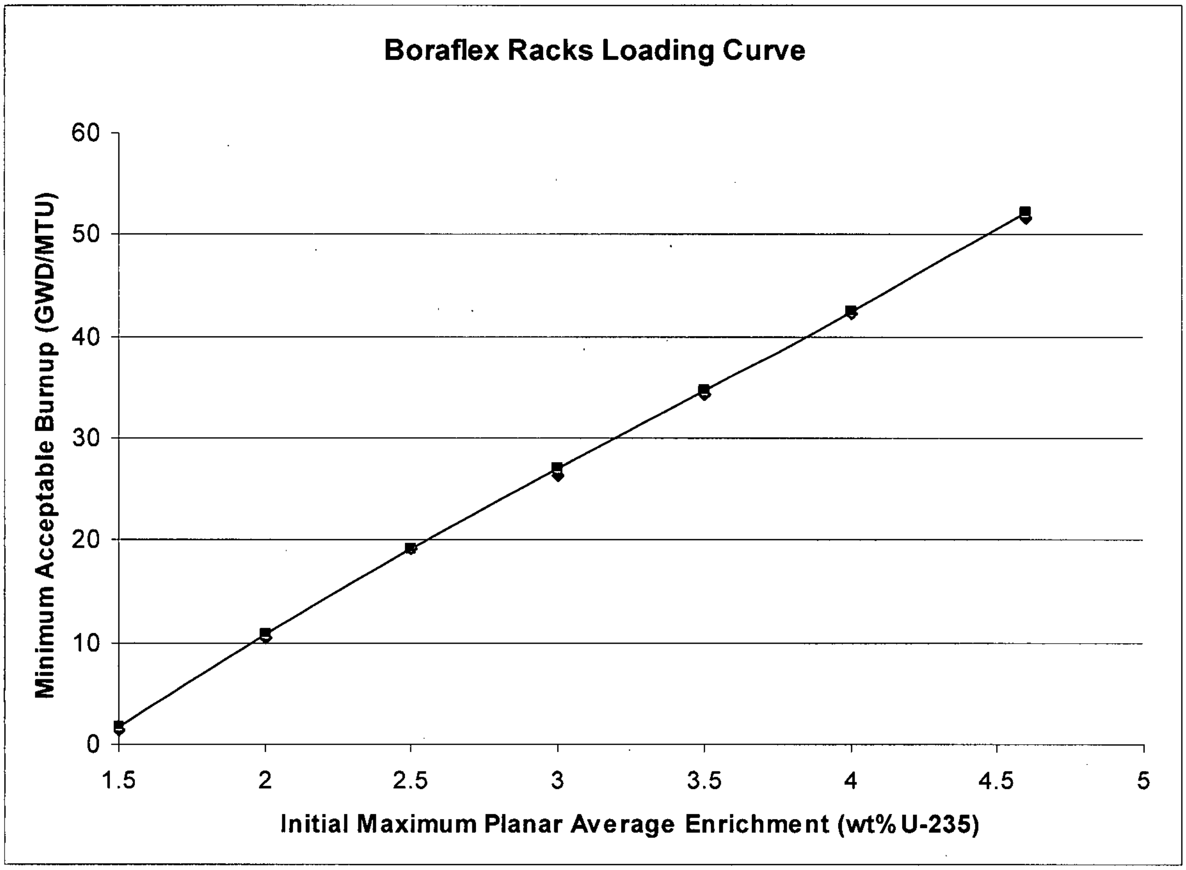

Fig. 2a: Minimum Fuel Burnup for Acceptable Storage of Spent Fuel of Various Initial Planar Average Enrichments (7 years).

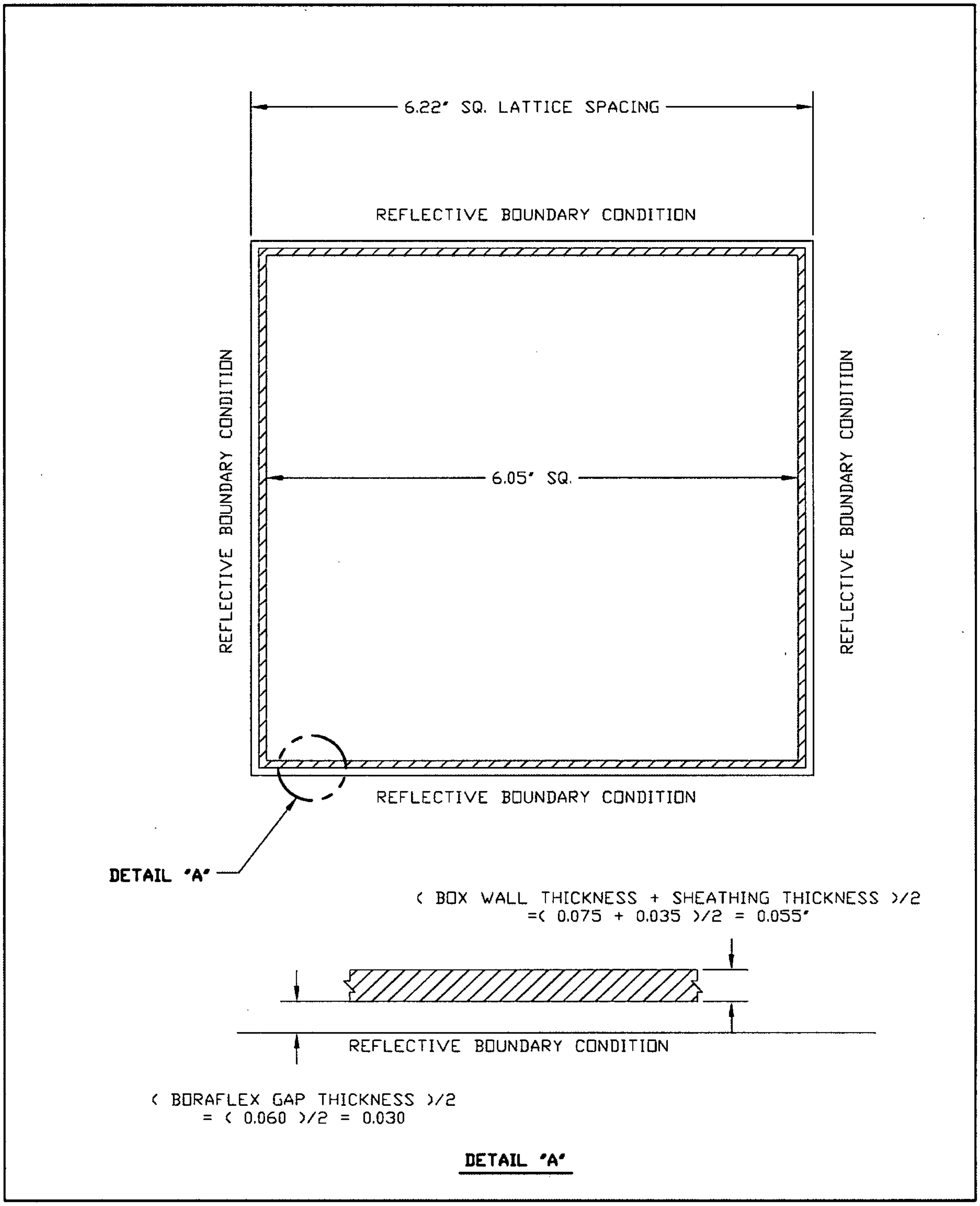

Fig. 3: Cross-Section of Typical Storage Cell (Calculational Model)