ENTERGY

SUMMARY OF THE WATERFORD 3 CRITICALITY SAFETY ANALYSES FOR FUEL ENRICHMENTS ABOVE 4.1 W/O U-235 TAKING CREDIT FOR FIXED BURNABLE POISONS

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1.0 SUMMARY

The Waterford 3 Spent Fuel Racks (SFR's) were originally designed for the storage of fuel containing up to 3.5 w/o U-235. The criticality safety analysis for these racks was later revised to support enrichments up to 4.1 w/o. Both of these analyses assumed no fixed poisons were present in the fuel assembly. Subsequently, a Boraflex monitoring program was proposed and approved by the NRC [Reference 1]. This program includes periodic non-destructive testing of selected Boraflex panels and criticality review. The initial testing and analysis review were successfully completed in 1993 [Reference 12].

With increasing cycle lengths, fuel management plans require a larger fraction of the reload batch to contain poisons. Because poison rods significantly reduce the reactivity of the fuel assembly, higher fuel enrichments may be loaded and still meet the fuel storage criticality acceptance criteria [Reference 11]. The Waterford 3 fuel design for poisoned assemblies contains fixed poison shims and does not permit the poison rods to be readily removed. Therefore, taking credit for the reactivity control effects of these poison rods is consistent with the criticality safety requirements [Reference 11].

This spent fuel pool criticality analysis demonstrates that, in the unborated SFR, a radially enrichment zoned assembly with enrichments of 4.50 and 4.10 w/o containing eight poison rods with 0.016 grams B-10 per inch (the "Base Assembly") meets the NRC acceptance criteria of 0.95 k-eff at the 95/95 probability/confidence level. Various other combinations of fuel enrichments and poison loadings were also analyzed and confirmed to meet the NRC acceptance criteria. These configurations include zoned assemblies with maximum pin enrichments up to 4.9 w/o U-235. Acceptable reactivity was achieved at a specific enrichment by varying the number of poison rods and/or concentration of poison material.

This report describes the KENO Monte Carlo calculations which model the Waterford 3 SFR for zoned and shimmed assemblies containing (1) the ABB CE Guardian Grid lower grid design, (2) increased enrichment above 4.1 w/o, and (3) and the presence of fixed burnable poison shims. The criticality analysis conservatively assumes that all Boraflex panels contain 4–1/2 inch coplanar gaps at the most reactive axial location (top of the panel). Uncertainties and biases due to methods and manufacturing tolerances are addressed in determining the 95/95 k-eff values for the assemblies.

The previous criticality analysis [Reference 13] considered the effects of various accident configurations, and concluded that, when credit is taken for a minimum soluble boron concentration in the SFR water, the NRC acceptance criteria is met. The new criticality analysis demonstrates that the Base Assembly is less reactive than the assembly assumed in the previous

analyses [Reference 13]. The SFR accident analyses remains within the acceptance criteria when potential gaps in the Boraflex panels are modeled.

The Containment Temporary Storage Racks (CTSR's) rely on assembly spacing to control reactivity. The Base Assembly design was determined to be less reactive than the assembly evaluated in the Reference 13 analysis. Therefore, the previous criticality analysis remains bounding for the Base Assembly design when placed in the CTSR under both normal and accident conditions.

The analysis of the new fuel storage racks is not addressed in this report.

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2. Report: "Boraflex Test Results and Evaluation ", EPRI TR-101986, Project 2813-04, February, 1993.

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12. "Revised Boraflex Surveillance Program of the Spent Fuel Pool Racks, Waterford Steam Electric Station, Unit 3", TAC No. M86006, Docket No. 50-382, August 23, 1993.

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3.0 METHODOLOGY

3.1 Model Description

The Waterford 3 spent fuel rack was designed by Wachter Associates, Inc.. The rack is composed of cells containing stainless steel partitions which provide a fuel assembly storage area and two poison insert areas per cell. The cells are oriented so that face adjacent assemblies are separated by a poison insert area. Poison inserts contain two Boraflex panels which are encapsulated in stainless steel cladding. The Boraflex panels are restricted from movement by indentations in the clad which apply continuous pressure over the entire length of the panel. The panels are arranged in a rectangular configuration which maintains a 1 inch flux trap between panels. This configuration is illustrated by Figure 3.1-3.

The panels were designed to be positioned approximately axially symmetric with the centerline of the active fuel column leaving a small portion of the fuel uncovered at the top and bottom of each panel. The fuel assembly design analyzed in this report has a slightly elevated fuel column because of the use of the debris resistant ABB CE Guardian Grid lower grid design . See Figure 3.1-1 for the relative axial positioning of the active fuel, shim, and Boraflex regions.

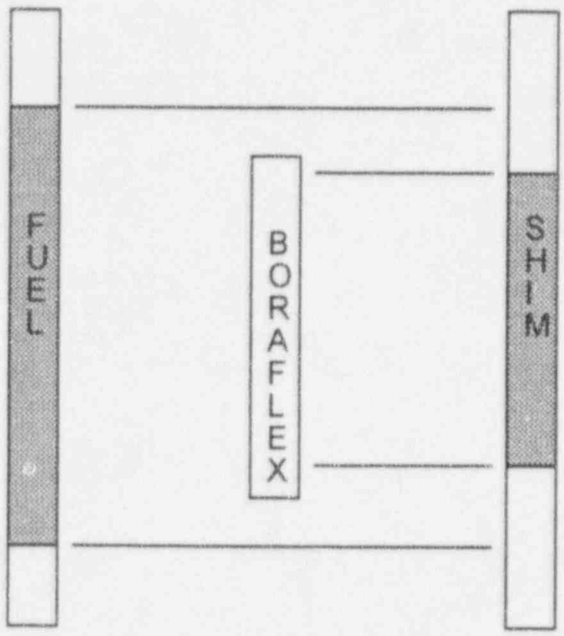
Waterford 3 uses an ABB CE 16x16 fuel assembly design with five large water holes for insertion of a control element assembly. The Base Assembly used in this analysis has an enrichment distribution containing 176 fuel pins at 4.50 w/o and 52 fuel pins at 4.10 w/o. The lower enrichment pins minimize power peaking around the five water holes and in the corners of the assembly. The position of the eight B4C burnable poison rods is shown in Figure 3.1-2. Other configurations of poison rods considered in this evaluation are also shown in that figure.

The analysis assumes that all fresh fuel assemblies with a maximum enrichment above 4.1 w/o U-235 contain boron shims which displace fuel rods. Neutron absorption by U-234 is credited and a bounding fuel stack density is used. The Boraflex is modeled using the minimum design dimensions and the 95/95 lower limit of the assayed B-10 loading. The panel width is also assumed to shrink 4.1 %, consistent with Reference 2. The panel height is reduced by the gap size as described below. Table 3.1-1 shows the key geometric parameters used in the base modeling of the fuel assembly and SFR. The assembly is positioned radially symmetric in the SFR cells which was determined to be the most reactive. The spent fuel pool water is modeled at maximum density and does not contain any soluble boron.

Formation of gaps occur when the stress in a restrained panel reaches the shearing strength of the Boraflex. A gap has three characteristics which effect the reactivity of the system; (1) gap size,

(2) axial position of the gaps, and (3) configuration of the gaps. Larger gap sizes result in higher system reactivity because the surface area of the absorber is reduced. The reactivity effect of axial position is dependent on two factors: (1) the interaction of a gap and the end of panel region, and (2) coupling of gaps in the same flux trap. If both panels in a flux trap form gaps, but the gaps occur at different axial locations, then the reactivity impact will be much lower than if the gaps are formed at the same location. The criticality analysis summarized in this report conservatively assumes that all Boraflex panels contain 4–1/2 inch coplanar gaps at the top of all of the panels (the most reactive axial position). This gap size bounds the Waterford 3 measurements described in Reference 1 and is consistent with the maximum gap size reported in Reference 2. This approach is very conservative relative to more realistic analyses which would include the effect of the variations in gap size and location based on probability of occurrence distributions. Inclusion of these effects would be expected to reduce the upper limit of the rack k-effective by 1-2% delta k.

Figure 3.1-1. Axial Geometry



Notes: The shim region of the poison rods is axially symmetric relative to the active fuel region.

The Boraflex Panels (without gaps) are shifted approximately 1 inch down relative to the center of active fuel. The top of the panel is less than 1 inch above the top of the shim region of the poison rods.

Figure 3.1-2. Assembly Geometry

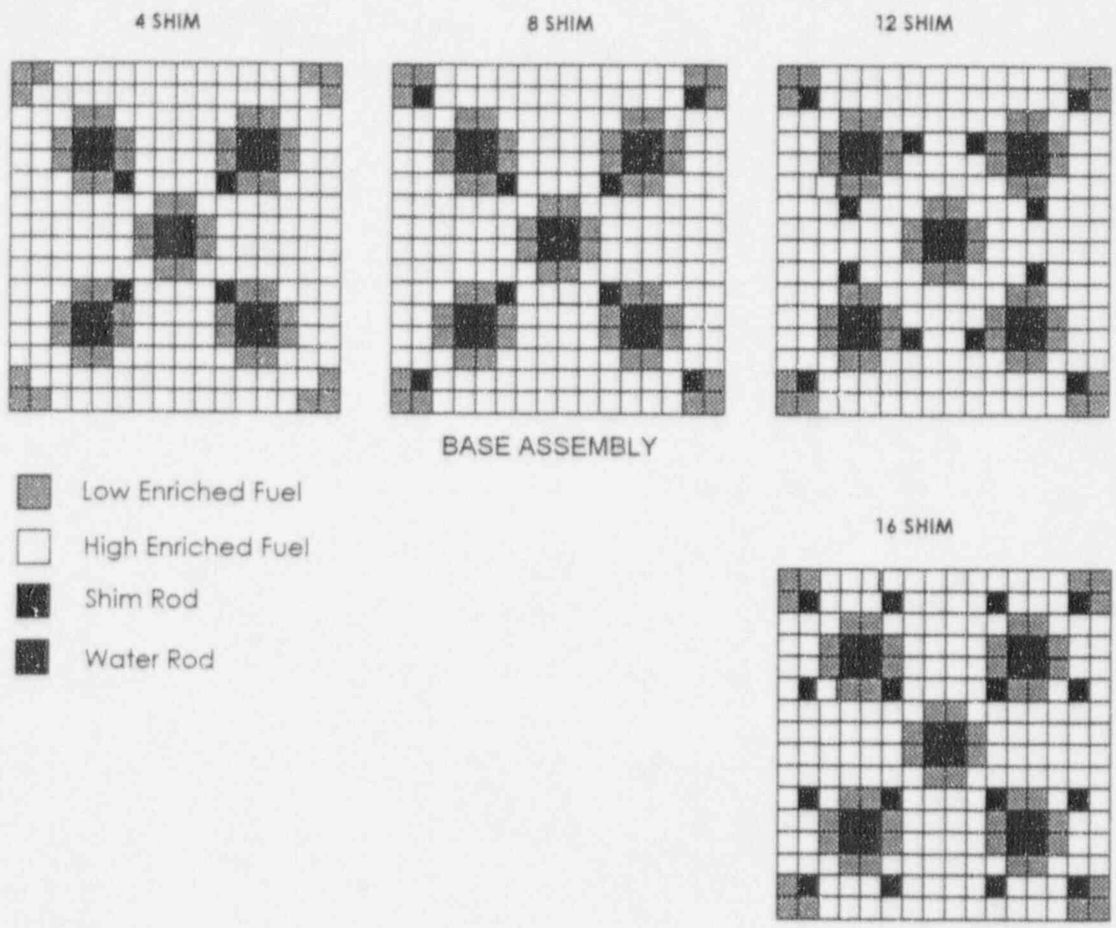
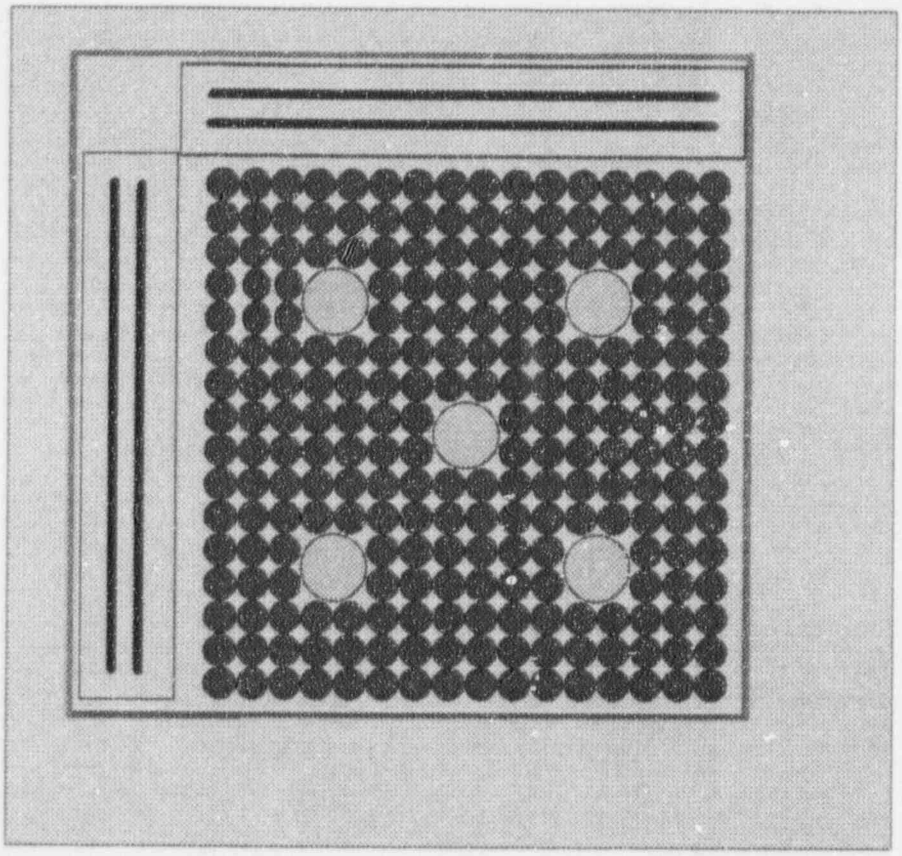


Figure 3.1-3. Radial Geometry of Analyzed Cell



Note : Periodic Boundary Conditions Applied to Unit Cell to Simulate an Infinite Radial Rack.

Table 3.1-1. WATERFORD 3 Dimensional Data

Parameter	Design Specification	Analyzed
Fuel Pellet diameter	0.325"	0.325"
Fuel Clad I.D.	0.332"	0.332"
Fuel Clad O.D.	0.382"	0.382"
Fuel Stack Density	10.06 gm/cc	10.2 gm/cc
Fuel Enrichments	4.05/3.65±0.005%	up to 4.90/4.50 w/o
Rod Pitch	0.506"	0.506"
Boraflex Panel Thickness	0.100+ 0.0-0.020"	0.080"
Boraflex Panel Height	139.0±0.5"	138.32"
Boraflex Panel Width	7.2+0.0-0.2"	6.713" (4.1% shrinkage)
Boraflex Areal Density	0.02299 gm B-10/cm^2	0.02259 gm B-10/cm^2
Insert Cladding Thickness	0.031"	0.031"
Insert Width	7.31"	7.31"
Flux Trap Thickness	1.0+0.1-0.0"	1.0"
Panel Centerline from SFR lower plate	79.7"	79.7"
Poison Area Thickness	1.57+0.02-0.0"	1.57"
Poison Area Width	8.59±0.03"	8.59"
SFR Stock Thickness	0.093±0.005"	0.093"
SFR Cell I.D.	10.164+0.060-0.0"	10.164"

3.2 Spent Fuel Rack

The analytical methods to determine SFR reactivity use the computer codes SCALE 4 [Reference 8] and CASMO 3 [Reference 3]. Both codes have been widely used for criticality analysis throughout the nuclear industry. While the SCALE code has been extensively benchmarked by the industry, specific benchmark comparisons to critical experiments were performed to establish bias and uncertainty factors. The details and results of this benchmarking are described in Appendix A of this report.

3.2.1 Base Analyses

As noted in the previous section of this report, the SCALE analyses conservatively assume that 4-1/2 inch gaps occur at the top of every panel. The SCALE calculations are run for 210 generations with 3000 histories per generation. As the first ten generations are skipped in these analyses, a total of 600,000 histories are analyzed.

3.2.2 Tolerance Factor Analyses

Uncertainties or tolerance factors in the rack and fuel design parameters are evaluated by either (1) setting the parameter to its most conservative value, or (2) performing sensitivity studies to determine the reactivity impact of the tolerance factor. These sensitivity studies were performed using the CASMO model [Reference 3].

The following tolerance factors were analyzed:

1. Higher fuel enrichments (nominal + 0.05 w/o U-235), calculated for increased fuel loading;
2. Higher fuel pellet density, calculated for increased fuel loading;
3. Higher form factor, calculated for increased fuel loading;
4. Larger fuel pellet diameter, calculated for increased fuel loading;
5. Larger clad I.D., calculated for reduced absorption by the fuel cladding;
6. Smaller clad O.D., calculated for increased neutron moderation in the fuel assembly;
7. Minimum guide tube thickness, calculated for increased neutron moderation in the fuel assembly;
8. Larger fuel pit area, calculated for increased neutron moderation around the fuel assembly;

9. Smaller shim lot loading, calculated for reduced neutron absorption in shims due to uncertainties in the average loading of a shim lot;

10. Smaller shim pellet loading, calculated for reduced neutron absorption in shims due uncertainties in the individual pellet loading relative to the average of a lot.

The base CASMO k-inf was subtracted from the calculated CASMO k-inf's from cases 1 through 10 above to obtain the delta-k values for each of the tolerances. The tolerance factors were then used to obtain the combined fabrication tolerance factors for each of the assembly designs. As these effects are independent, they were statistically combined by taking the square root of the sum of the squares of each contributing uncertainty due to the tolerances identified above.

3.2.3 Resultant 95/95 Probability/Confidence Reactivities

The SCALE calculated eigenvalue for each analyzed geometry was combined with the appropriate tolerance uncertainties and method uncertainties as follows:

$$k_{eff\text{-}95/95} = k_{SCALE} + \Delta k_{tolerances} + \Delta k_{enrichment} + \Delta k_{method} + \kappa^{*}\sigma_{total}$$

where:

$\Delta k_{tolerances}$ = overall tolerance uncertainty

$\Delta k_{enrichment}$ = 1.0 - (1.001204 - 0.001894*w/o) from Appendix A

Δk_{method} = 0.00072 from Appendix A

σ_{total} = $(\sigma_{SCALE}^2 + \sigma_{method}^2)^{1/2}$

f = $[\sigma_{total}^4/(\sigma_{SCALE}^4/(210-10-1) + \sigma_{method}^4/(21-1))] - 2$

Note: This formulation was taken from Reference 7.

Note: 210 generations (skipping 10 generations)

and 21 critical experiments were analyzed.

κ = 95/95 tolerance factor for f degrees of freedom

3.2.4 Modeling Conservatisms

The following modeling conservatisms have been identified:

1. The nominal Boraflex dimensions used are the minimum as-designed values with an additional conservative 4.1 % width shrinkage and a 4–1/2 inch gap assumed at the top of each panel.
2. Worst case geometry (symmetric) is used for the positioning of the fuel assembly in the SFR pit.
3. Water is set to the maximum density of 1 gm/cc.
4. The infinite lattice Dancoff factor is used.
5. No credit is taken for fuel burnup.
6. No credit is taken for the soluble boron in the water (except for accidents).
7. Much structural steel is not modeled in the problem.
8. No neutron leakage exists in the x-y directions.

3.3 Containment Temporary Storage Rack

The containment temporary storage rack (CTSR) relies upon assembly spacing to control reactivity for normal operations and credits the presence of soluble boron for accident conditions. The rack was previously analyzed as documented in Reference 13. The continued applicability of that analysis was evaluated by comparing the reactivity of the assembly design assumed in that analysis to the Base Assembly design, using the SCALE code for both designs. Two assemblies, using the Base Assembly design, were modeled assuming pure maximum density water with separation distances ranging from 1.762 to 12 inches. This range includes state points consistent with nominal operating configurations and the minimum separation distance for a postulated fuel assembly handling accident. The same calculations were repeated using the Reference 13 design.

3.4 Fuel Handling Accidents

The Reference 13 criticality analysis considered the effects of various fuel handling accident configurations. This analysis demonstrated significant margin to the 0.95 acceptance criteria when credit for boron was considered. The applicability of that analysis was evaluated by comparison of the reactivity of the Base Assembly design relative to that assumed in the Reference 13 analysis. This evaluation considered the effects of variations in separation distance as described in Section 3.3 above and the effects due to the presence of the strong absorbers. It also considered the impact of Boraflex gaps on the spent fuel pool accident analysis. These evaluations were performed using the SCALE code.

4.0 RESULTS

4.1 Spent Fuel Rack Results

4.1.1 Reactivity vs. Axial Gap Location

An evaluation was performed to determine the most reactive axial location to place the coplanar gaps in the Boraflex. The results of placing coplanar gaps at various positions for the Base Assembly (4.5/4.1 w/o zoned fuel with 8 shims at 0.016 grams B-10/inch) are shown in Figure 4.3-1. As can be seen, the most reactive placement was found to be at the top of the panel.

4.1.2 Tolerance Results

As described in Section 3.2.2, the base CASMO reactivities were subtracted from the calculated CASMO reactivities for the off-nominal cases to obtain the delta-k values for each of the tolerances. The tolerance factors were then used to obtain the combined fabrication tolerance factors for each of the assembly designs. As the tolerances are independent, the statistically combined tolerance factors (Table 4.3-1) were obtained by taking the square root of the sum of the squares of each contributing uncertainty due to the tolerances identified above.

4.1.3 Final Reactivities

The raw KENO eigenvalues and uncertainties were combined with the tolerance uncertainties and the methods uncertainties to obtain the final 95/95 values for k-eff. The components and the final 95/95 k-eff's are given in Table 4.3-2. These results demonstrate that the k-effective is less than 0.950, including uncertainties and biases, at the 95/95 probability/confidence level. Therefore, the acceptance criteria of k-effective below 0.95 is met for the Base Assembly and other equivalent reactivity designs. A plot of the reactivity equivalent designs, i.e., those combinations of fuel enrichment and shim loadings which give final reactivities less than 0.95, is given in Figure 4.3-2.

4.2 Containment Temporary Storage Rack Results

The Reference 13 assembly design was found to bound the Base Assembly design for all configurations. Since the assessment considers tightly coupled configurations equivalent to accident conditions and loosely coupled configurations equivalent to normal operating conditions the Reference 13 results are bounding. Those results demonstrate that k-effective is less than 0.899 for normal operations and less than 0.90 for accident conditions including uncertainties and biases at the 95/95 probability/confidence level. Therefore, the acceptance criteria of k-effective below 0.95 is met for the Base Assembly and other equivalent reactivity designs.

4.3 Accident Results

The reactivity of the Reference 13 assembly design was found to bound the reactivity of the Base Assembly design. The positive reactivity impact due to the presence of gaps was conservatively determined to be 0.05 delta-k. When this value added to the Reference 13 analysis results, the k-effective is less than 0.91, including uncertainties, when credit for soluble boron is included. Therefore, the acceptance criteria of k-effective below 0.95 is met for the Base Assembly and other equivalent reactivity designs.

Table 4.3-1. Tolerances

FUEL	STD	STD	STD	STD	STD	STD	STD	STD	STD	STD	STD
ENRICH.	4.11	4.2	4.24	4.33	4.37	4.5	4.55	4.61	4.65	4.85	4.9
# SHIMS	4	4	4	4	8	8	8	8	8	12	16
B10-/IN.	0.012	0.016	0.02	0.024	0.012	0.016	0.02	0.024	0.028	0.02	0.012
DELTA 1	0.00241	0.00234	0.00232	0.00225	0.00223	0.00215	[illegible]	0.00208	0.00204	0.00203	0.00202
2	0.00057	0.00057	0.00058	0.00058	0.00054	0.00059	0.00058	0.00059	0.00059	0.00064	0.00065
3	0.00153	0.00153	0.00155	0.00153	0.00152	0.00156	0.00156	0.00157	0.00157	0.00169	0.00172
4	0.00038	0.00038	0.00039	0.00039	0.00035	0.00039	0.00040	0.00040	0.00039	0.00043	0.00045
5	0.00001	0.00000	0.00002	0.00001	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
6	0.00238	0.00238	0.00247	0.00236	0.00234	0.00240	0.00242	0.00242	0.00243	0.00234	0.00232
7	0.00095	0.00090	0.00103	0.00090	0.00094	0.00095	0.00094	0.00099	0.00100	0.00108	0.00105
8	0.00108	0.00095	0.00104	0.00104	0.00084	0.00083	0.00082	0.00080	0.00079	0.00086	0.00082
9	0.00185	0.00157	0.00163	0.00150	0.00200	0.00203	0.00202	0.00195	0.00194	0.00299	0.00361
10	0.00093	0.00092	0.00099	0.00090	0.00116	0.00120	0.00118	0.00113	0.00113	0.00175	0.00213
RMS	0.00454	0.00436	0.00449	0.00429	0.00448	0.00452	0.00449	0.00446	0.00444	0.00519	0.00567

Note: DELTA = delta k, tolerance - nominal

Table 4.3-2. Final Reactivities

#SHIMS	4	4	4	4	8	8	8	8	8	12	16
GM B-10/IN	0.012	0.016	0.02	0.024	0.012	0.016	0.02	0.024	0.028	0.02	0.012
ENRICHMENT	4.11/3.71	4.20/3.80	4.24/3.84	4.33/3.93	4.37/3.97	4.50/4.10	4.55/4.15	4.61/4.21	4.65/4.25	4.85/4.45	4.90/4.50
KENO KEFF	0.92904	0.92958	0.93141	0.93285	0.93004	**0.93290**	0.93033	0.93109	0.93065	0.93148	0.92731
BIASES											
METHOD	0.00072	0.00072	0.00072	0.00072	0.00072	**0.00072**	0.00072	0.00072	0.00072	0.00072	0.00072
ENRICH.	0.00658	0.00675	0.00683	0.00700	0.00707	**0.00732**	0.00741	0.00753	0.00760	0.00798	0.00808
UNCERT.											
KENO	0.00109	0.00120	0.00124	0.00130	0.00118	**0.00120**	0.00122	0.00116	0.00125	0.00123	0.00135
METHOD	0.00156	0.00156	0.00156	0.00156	0.00156	**0.00156**	0.00156	0.00156	0.00156	0.00156	0.00156
TOLER	0.00454	0.00436	0.00449	0.00429	0.00448	**0.00452**	0.00449	0.00446	0.00444	0.00519	0.00567
TOTAL	0.00190	0.00197	0.00199	0.00205	0.00196	**0.00197**	0.00198	0.00194	0.00200	0.00199	0.00206
f	40.980	47.133	48.917	54.884	46.250	**47.133**	48.022	44.408	49.883	48.979	55.569
k	2.11802	2.07978	2.07016	2.04027	2084734	**2.07978**	2.07482	2.09557	2.06514	2.06983	2.03691
k*TOTAL	0.00402	0.00410	0.00412	0.00418	0.00409	**0.00410**	0.00411	0.00407	0.00413	0.00412	0.00420
95/95 K-EFF	0.94490	0.94551	0.94757	0.94904	0.94640	**0.94956**	0.94706	0.94787	0.94754	0.94949	0.94598

Figure 4.3-1. KENO K-eff for Eight Coplanar Gaps

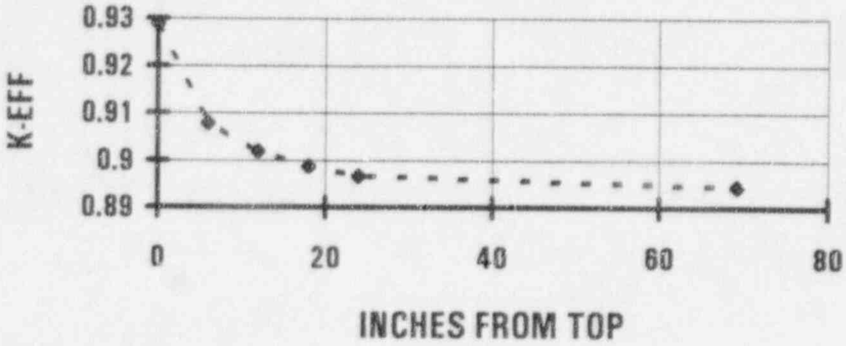
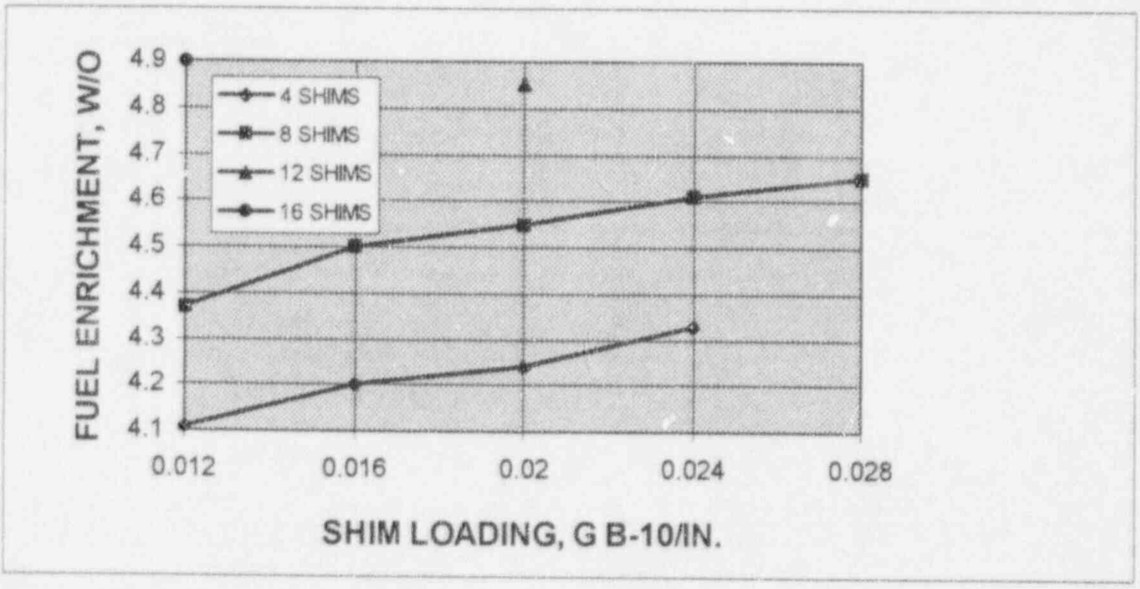


Figure 4.3-2. Reactivity Equivalent Designs



APPENDIX A: BENCHMARK OF SCALE4

A benchmarking of the SCALE 4 system of codes against critical experiments is performed in order to qualify and validate its use for performing fuel storage rack criticality safety analyses at Entergy Operations, Inc. plants.

The benchmarked SCALE 4 codes includes the CSAS25 Control Sequence of the ORNL Criticality Safety Analysis Sequence No. 4 (CSAS4). The functional modules sequentially executed by the CSAS25 control module are the BONAMI-S code, the NITAWL-S code, and the KENO V.a code [Reference 8].

A total of twenty-one critical experiments are used to benchmark these codes. These criticals are selected from a list of seventy-five critical experiments conducted by Babcock and Wilcox and Pacific Northwest Laboratory [Reference 4, Reference 5, and Reference 9]. The twenty-one criticals are chosen because of their fuel characteristics, lattice geometry's water gap spacing and materials are reasonably representative of those found in Entergy Operations, Inc. fuel storage rack arrays.

An evaluation of the benchmarking results identifies two significant biases in the SCALE calculated k-eff's. The first is an observed trend toward over-prediction of reactivity and increasing Boron loading in the Boral plates for four of the Babcock and Wilcox cores, as reported also in Reference 10. These four cases are corrected first for the Boron loading bias before further data reduction is performed. For conservatism, this credit is not taken in analyzing fuel storage rack.

The second observed bias is an under-prediction of reactivity with increasing fuel enrichment as follows:

Enrichment Bias = 1.0 - (1.001204 - 0.001894 * (Enr. in wt% U-235))

Statistical analysis of the 21 calculated k-eff's, using Criterion 2 [Reference 6], gives an enrichment corrected mean k-eff of 0.99915 with a Monte Carlo uncertainty of ±0.00305, a method bias of 0.00072, and a method uncertainty of ±0.00156. Table A.0-1 is a summary of the benchmark statistics.

Table A.0-1. Summary of SCALE4 Benchmark

Case No.	Case Name	Enrichment wt% U235	SCALE K-eff	Standard Deviation	Corrected K-eff	No. of Gen's	No. of Neutrons per Gen
1	BWII	2.46	0.99680	0.00238	0.99825	100	500
2	BWIII	2.46	0.99452	0.00285	0.99697	100	500
3	BWXI	2.46	0.99324	0.00245	0.99569	100	500
4	BWXIII	2.46	1.01005	0.00335	1.00004	100	500
5	BWXIV	2.46	1.00550	0.00293	0.99935	100	500
6	BWXVII	2.46	0.99950	0.00286	1.00431	100	500
7	BWXIX	2.46	0.98885	0.00262	0.99619	100	500
8	PNL5	2.35	0.98199	0.00298	0.99418	100	500
9	PNL26	2.35	1.00232	0.00287	1.00451	100	500
10	PNL28	2.35	0.99807	0.00283	1.00026	100	500
11	PNL32	2.35	0.99242	0.00325	0.99461	100	500
12	PNL33	2.35	1.00164	0.00337	1.00383	100	500
13	PNL38	2.35	0.99832	0.00265	1.00051	100	500
14	PNL39	2.35	0.99364	0.00330	0.99583	100	500
15	PNL8	4.29	0.99663	0.00323	1.00349	100	500
16	PNL9	4.29	0.98640	0.00321	0.99326	100	500
17	PNL10R	4.29	0.99441	0.00338	1.00127	100	500
18	PNL11	4.29	0.99480	0.00354	1.00166	100	500
19	PNL12	4.29	0.99321	0.00344	1.00007	100	500
20	PNL13	4.29	0.99039	0.00297	0.99725	100	500
21	PNL32	4.29	0.99587	0.00329	1.00273	100	500

***Avg. Corrected K-eff* = 0.99615**
***Monte Carlo Uncertainty* = 0.003053**
***Method Bias* = 0.000721**
***Method Uncertainty* = 0.001559**

NPF-38-163

ATTACHMENT IV

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 23, 1993

Docket No. 50-382

RECEIVED
[redacted]
AUG 27 1993
ILN: 93-0180

Mr. Ross P. Barkhurst
Vice President Operations
Entergy Operations, Inc.
Post Office Box B
Killona, Louisiana 70066

Dear Mr. Barkhurst:

SUBJECT: REVISED BORAFLEX SURVEILLANCE PROGRAM OF THE SPENT FUEL POOL RACKS WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. M86006)

By letter dated December 31, 1992, the staff of Entergy, Inc. (the current licensee, formerly Louisiana Power and Light [LPL]), submitted a revised surveillance program for monitoring the Boraflex panels contained in the spent fuel racks at the Waterford Steam Electric Station, Unit No. 3 (Waterford 3). It had been determined earlier that use of coupons might not be adequate to monitor the extent of gamma radiation induced gaps in spent fuel pool Boraflex panels. Consequently, in December 1987, LP&L proposed an alternative surveillance program for the Boralflex panels in lieu of the previous Boraflex coupon surveillance program. The new surveillance program, as documented in Section D of the attachment to the December 31, 1992, letter, includes the following steps:

- Gamma exposure tracking of the spent fuel pool Boraflex panels.

- Periodic nondestructive testing, at a maximum 4-year interval time, of selected Boraflex panels. The testing uses neutron attenuation (blackness testing) as a means of detecting gaps (discontinuities) along the Boraflex panels. The licensee's nondestructive testing data of selected Boralflex panels in November 1992 will be used to provide a baseline for the trending of gap formation in the panels.

- Sixteen spent fuel pool exposure cells with freshly discharged fuel in them to serve as samples for trending the effects of high gamma radiation exposure on Boraflex panels. These 16 exposure cells will be tested as described above and are expected to lead the spent fuel storage racks in gap formation.

- Periodic destructive testing on selected panels if engineering assessment determines it is necessary.

- Monitoring and comparisons to industry developments to determine the latest methods for testing and monitoring Boraflex performance.

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We have reviewed the alternative Boralflex surveillance program and find that the program provides an acceptable means of monitoring the integrity of the Boraflex panels used in the construction of the Waterford 3 spent fuel pool storage racks. The baseline surveillance results in the submittal indicate that the Boraflex panels in the spent fuel pool storage racks are currently capable of performing their intended safety function.

The Entergy submittal of December 31, 1992, indicates that you have met your prior commitments to (1) terminate the Boraflex coupon surveillance program, (2) establish and maintain a database of the calculated Boraflex panel gamma exposures in the spent fuel storage racks, (3) submit nondestructive test results on several Boraflex panels, and (4) establish a surveillance program as discussed above. We agree that you have completed the first three commitments and that you will continue to have a surveillance program as necessary to meet the stated objectives.

If you have any questions on this matter, please let me know. This completes the action under TAC No. M86006.

Sincerely,

DLWigginton

David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

cc: See next page

Mr. Ross P. Barkhurst
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NPF-38-163

ATTACHMENT V

W3F192-0396
A4.05
QA

December 31, 1992

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Boraflex Surveillance Program

Gentlemen:

Waterford 3 committed in December 1987 to propose to the NRC by January 1, 1993 an appropriate surveillance program for surveillance of Boraflex in the spent fuel storage racks. The purpose of this letter is to provide to the NRC the information which satisfies this commitment including a discussion of the detailed commitments and the description of the proposed surveillance program. This information is provided in the attachment to this letter.

Please contact me or Robert J. Murillo should there be any questions regarding this letter.

Very truly yours,

R.F. Burski
Director, Nuclear Safety

RFB/RJM/dc
Attachment
cc: J.L. Milhoan, NRC Region IV
D.L. Wigginton, NRC-NRR
R.B. McGehee
N.S. Reynolds
NRC Resident Inspectors Office

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Attachment to Letter W3F192-0396

A. Background

The pertinent information regarding the Boraflex surveillance program was documented in the following documents: Louisiana Power and Light (LP&L) letter W3P87-2055 dated 9/16/87, LP&L Letter W3P87-2527 dated 12/15/87, and NRC Safety Evaluation Report (SER) dated 12/21/87. These documents established NRC and Waterford 3 agreement for the following commitments:

1. The current Boraflex coupon surveillance commitment will not be performed.

2. Waterford 3 will develop a log to track the gamma dose buildup in the spent fuel storage racks (SFSR).

3. Waterford 3 will provide the NRC by January 1, 1993 with actual data, from nondestructive techniques, on several Boraflex panels to verify that the poison material has not been unacceptably degraded by the formation of gaps.

4. Waterford 3 will provide the NRC by January 1, 1993 with a surveillance program to verify the effectiveness of the Boraflex poison material in the Waterford 3 spent fuel storage racks. This surveillance program will be based on the latest industry developments on Boraflex surveillance methods and techniques as well as studies on Boraflex degradation mechanisms from radiation exposure.

B. Commitment Resolution

The resolution for each of the foregoing commitments is the following:

1. The Boraflex coupon surveillance program was terminated and is not active.

2. Waterford 3 maintains a database of the calculated Boraflex panel gamma exposures in the Waterford 3 SFSR.

3. Testing of Boraflex panels was performed in November, 1992. A review of the test results indicates that the Boraflex continues to perform its intended neutron attenuation function. Additional information is provided in section C of this attachment.

4. The proposed surveillance program has been developed, and it is described in section D of this attachment.

C. Testing of Boraflex Panels

Testing of a representative sample of Boraflex panels was performed in November, 1992. A review of the test results indicates that the Boraflex continues to perform its intended neutron attenuation function.

Testing was conducted to test the highest exposed panels for gaps. The test results are therefore bounding on expected gap formation in the Boraflex.

The following is a summary of the test results:

Number of Panels Tested	=	697
Total Number Of Gaps	=	185
Number of Panels With No Gap	=	538
Average Exposure	=	8.03E+09 rads
Maximum Exposure	=	1.66E+10 rads
Number of Panels With 1 Gap	=	136
Average Exposure	=	8.93E+09 rads
Maximum Exposure	=	1.66E+10 rads
Average Gap Size (A&B)	=	2.17 inches
Maximum Gap Size (A&B)	=	3.57 inches
Number of Panels With 2 Gaps		20
Average Exposure	=	9.00E+09 rads
Maximum Exposure	=	1.62E+10 rads
Average Gap Size (A&B)	=	2.04 inches
Maximum Gap Size (A&B)	=	2.74 inches
Number of Panels With 3 Gaps	=	3
Average Exposure	=	1.01E+10 rads
Maximum Exposure	=	1.31E+10 rads
Average Gap Size (A&B)	=	1.05 inches
Maximum Gap Size (A&B)	=	1.24 inches

The distribution of gaps size indicates that the Boraflex is performing as expected based on EPRI/industry data. EPRI data indicates that after approximately 7x10E+09 rads, shrinkage, and gap formation, of the panels reaches a maximum value.

We expect that the locations with fuel currently stored will have a much lower number of gaps, due to being in lower exposed areas.

D. Description of Boraflex Surveillance Program

Objective

The objectives of the Boraflex surveillance program are:

1. To verify the effectiveness of the Boraflex poison material in the SFSR.

2. To ensure the requirements of Technical Specification 5.6.1 are met. These requirements are:

   The spent fuel storage racks are designed and shall be maintained with:

   a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties.

   b. A nominal 10.38 center-to-center distance between fuel assemblies placed in the spent fuel storage racks.

3. To ascertain the rate of change (gap formation) in the Boraflex and determine the interval between surveillances.

Methodology

The Boraflex surveillance program at Waterford 3 will consist of the following:

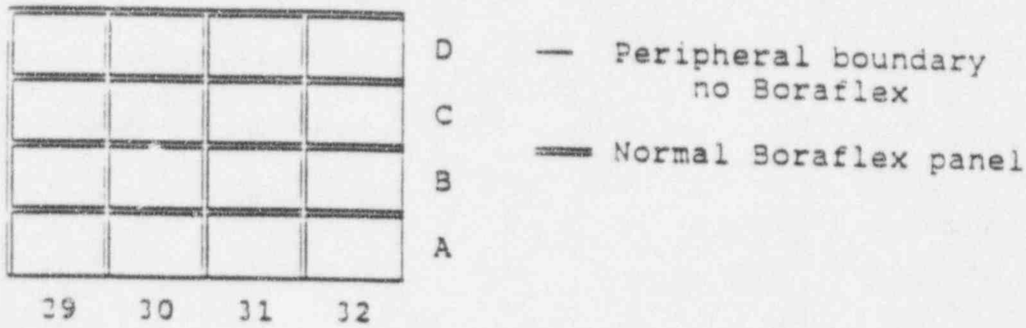
1. Tracking of the gamma exposure of the Boraflex panels.

2. Periodic nondestructive testing ("blackness" testing) of selected Boraflex panels to determine the extent of gap formation. The proposed schedule for the blackness testing is four (4) year intervals but no later than 12/31/94 for the next testing with the exact schedule being determined by the rate of gap formation.

   The results of the nondestructive testing will be compared to industry data and EPRI sponsored research. The Waterford 3 SFSR criticality analysis will be reviewed by April 15, 1993 based on EPRI data for maximum exnected gap formation. The review will be performed by modeling the current status of the Boraflex panels and projecting the expected gap formation due to calculated exposure through 12/31/94. Data collected in November, 1992 will provide a baseline for the trending of gap formation.

Waterford 3 will have lead exposure cells in the SFSR to ensure that degradation due to gamma exposure will be identified early. These 16 cells will have freshly discharged fuel loaded into them following refueling outages, and these cells will be tested during each nondestructive test. The configuration of these cells in the SFSR is shown in Figure One (1).

FIGURE ONE



Since these locations contain some of the leading exposure panels, fuel will not be placed into these cells until the review of the criticality analysis is complete. These lead exposure cells will provide data for the performance of Boraflex under gamma exposure and early indication of unacceptable trends. These cells are expected to lead the SFSR in gap formation.

3. Periodic destructive testing of selected panels if engineering assessment determines a need for destructive testing. The need for destructive testing will be determined by a review of nondestructive test results, trend data, and industry experience.

4. Monitoring of industry developments to determine the latest methods for testing Boraflex and to determine Boraflex performance at other sites.

Summary

The Boraflex surveillance program will provide sufficient data to verify the effectiveness of the Boraflex poison material and to review the SFSR criticality analysis. Additionally, it will provide trending data for changes in the Boraflex with gamma exposure and thus provide a technical basis for taking early corrective action if problems arise. The surveillance program satisfies the program objectives and provides assurance that the margin of safety in the SFSR will be maintained.

W3F192-0396

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