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Enclosure 6

AREVA Report ANP-2642(NP), Revision 0,
Brunswick Nuclear Plant
Spent Fuel Storage Pool Criticality
Safety Analysis for ATRIUM-10 Fuel,
dated September 2007

An AREVA and Siemens company

ANP-2642(NP)
Revision 0

Brunswick Nuclear Plant Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM-10 Fuel

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial release.

Contents

Tables

Figures

This document contains a total of 33 pages.

Nomenclature

BOL	beginning of life
BWR	boiling-water reactor
GWd	energy unit, giga-watt-day
k-eff	effective neutron multiplication factor
k_∞	infinite lattice neutron multiplication factor
PLR	part-length fuel rod
PWR	pressurized water reactor
NRC	Nuclear Regulatory Commission, U. S.
REBOL	reactivity-equivalent at beginning of life
TD	theoretical density

1.0 Introduction

This report presents the results of a criticality safety evaluation performed for the storage of AREVA NP Inc.* ATRIUM™-10† fuel assemblies in the BWR spent fuel storage racks of the Brunswick Nuclear Plant, (circular tube and square tube with BORAL). Information on these spent fuel storage arrays is presented in References 1, 2, and 3. The previous analyses for the fuel in the Brunswick Nuclear Plant spent fuel storage pool are documented in the information provided in References 1 and 2.

Results included in this report define a reference ATRIUM-10 lattice design that bounds current lattice designs at extended power uprate conditions. With the reference design stored in the spent fuel storage pool, the k-eff will remain below the NRC acceptance criterion (worst case array k-eff ≤ 0.95).

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* AREVA NP Inc. is and AREVA and Siemens company.

† ATRIUM is a trademark of AREVA NP.

2.0 Summary

The ATRIUM-10 fuel design can be safely stored in the Brunswick Nuclear Plant spent fuel storage racks (see Reference 3) per the criticality safety limits defined in Table 2.1. The fuel defined by a bounding reference lattice of 4.5 wt% U-235 with 8 gadolinia rods at 4.0 wt% Gd_2O_3 (AT10-450L-8G4 where L refers to lattice enrichment and G refers to gadolinia loading) does not exceed an array k-eff (under worst credible conditions) of 0.95 in the Brunswick Nuclear Plant spent fuel storage pool. Fuel that has less enrichment and higher gadolinia content in each lattice than the reference AT10-450L-8G4 lattice may be stored and will not exceed a k-eff of 0.95.

ATRIUM-10 fuel that does not conform to the bounding reference fuel lattice conditions can be analyzed for storage in the spent fuel pool racks by using the CASMO-4 sample input presented in Appendix A. If the cold in-rack k_∞ for this fuel is less than that defined in Table 2.1, then it can be safely stored in the Brunswick Nuclear Plant spent fuel racks.

The spent fuel storage facility for the Brunswick plants contain two types of BWR storage racks.

1) a low density, circular tube fuel storage rack

2) a high density Boral storage rack.

Calculations confirm that the high density fuel storage racks are more reactive and therefore bound the circular tube storage rack. Hence, only the high density storage racks will be addressed on a quantitative basis in this report.

The KENO evaluation summarized in this report includes manufacturing uncertainties for the ATRIUM-10 fuel design and the fuel pool storage racks, code modeling uncertainties, reactivity increases due to accident conditions, and an uncertainty multiplier to determine the 95/95 upper limit k-eff. The conditions and uncertainties assumed in this analysis are described in Section 6.0.

This analysis considers unchanneled fuel assemblies as well as assemblies with the AREVA advanced channel or the 100 mil fuel channel. Additionally, there is no limitation for bundle orientation or position of the bundle in the storage cell since these are accounted for in the analyses.

To assure that the actual reactivity will always be less than the calculated reactivity, the following conservative assumptions have been made:

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Table 2.1 Criticality Safety Limits for ATRIUM-10 Fuel Assemblies Stored in the Brunswick Nuclear Plant Spent Fuel Storage Pool

1. ATRIUM-10 Fuel Configuration

Parameter	Nominal ATRIUM-10 Value
Clad OD, in.	0.3957
Clad ID, in.	0.3480
Pellet Diameter, in.	0.3413
Rod Pitch, in.	0.510
Fuel Density % Theoretical	95.85 to 96.26*
Water Rods	Internal Channel

2. Fuel Design Limitations†

Maximum Lattice Average Enrichment, wt% U-235	4.50
Minimum Number Gd‡ Rods	8
Minimum wt% Gd_2O_3 in each Gd Rod	4.0

Fuel may be stored with or without fuel channels.

3. ATRIUM-10 fuel assemblies which do not meet the limitations above may be stored in the BWR storage racks provided the reactivity of any axial lattice in the assembly does not exceed the CASMO-4 k_∞ (in-rack) of **0.880** at any point in its lifetime. (The CASMO-4 in-rack geometry to be used for this calculation is defined in Appendix A. The calculation is run with CASMO-4 version UMAR06 at xenon-free conditions with fuel and moderator temperatures of 4°C and a 100 mil fuel channel).

4. Spent fuel storage rack design parameters and dimensions are as defined in Reference 3.

* There is no significant difference (<1 mk) in array reactivity between 95.85% and 96.26% TD. Hence, fuel pellet density within this range is acceptable.

† All axial lattices of the assembly shall meet the stated limits, except natural blankets where the gadolinia limitations do not apply.

‡ Gd means gadolinia-bearing (Gd_2O_3).

3.0 Criticality Safety Design Criteria

The criticality safety design criteria defined in the following documents are assumed to be applicable for the Brunswick Nuclear Plant spent fuel storage facility evaluation and are consistent with the Brunswick FSAR and Technical Specifications:

A. Section 9.1.2 (Spent Fuel Storage) of the *Standard Review Plan* (Reference 4).

B. Regulatory Guide 1.13 (*Spent Fuel Storage Facility Design Basis*) issued by the NRC (Reference 5).

C. ANSI/ANS American National Standard 57.2-1983 (*Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants*) issued by the American Nuclear Society (Reference 6).

D. ANSI/ANS American National Standard 8.17-1984 (*Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors*) issued by the American Nuclear Society, January 1984 (Reference 7).

E. "OT Position for the Review and Acceptance of Spent Fuel Storage and Handling Applications," issued by the NRC in 1978 (Reference 8).

F. "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," issued by the NRC in 1998 (Reference 9).

These documents define the assumptions and acceptance criteria used in this evaluation. In descending order (from A to F), these documents go from "least" to "most" detail relative to explicitly defining what needs to be addressed in the criticality safety evaluation. In general, the criticality safety acceptance criterion applicable to this evaluation is as defined by Section 9.1.2 of the Standard Review Plan (Reference 4):

> The k-eff of the array shall not exceed 0.95 under worst credible storage array conditions,* including uncertainties,† or under accident conditions, e.g., dropped or misplaced assembly.

* Worst credible conditions include minimum storage rack dimensions, worst fuel assembly position in racks, and peak reactivity fuel at most reactive pool moderator temperature conditions, including boiling.

† Uncertainties include allowances for statistical uncertainty associated with the analytical method and benchmark calculation uncertainties.

4.0 Fuel and Storage Array Description

4.1 *Fuel Assembly Design*

The ATRIUM-10 fuel assembly is a 10x10 fuel rod array with an internal square water channel offset in the center of the assembly (taking the place of nine fuel rod locations). The assembly contains part-length fuel rods (PLR); therefore, a "top" lattice geometry will apply above the PLR boundary and a "bottom" lattice geometry will apply below the PLR boundary. The ATRIUM-10 mechanical design parameters are summarized in Table 4.1. A representation of the ATRIUM-10 assembly design is depicted in Figure 4.1. It is anticipated that the ATRIUM-10 fuel in the Brunswick Nuclear Plant will use the AREVA Advanced Channel with a thick corner and thin wall channel design.

4.2 *Low Density Fuel Storage Rack*

This storage rack is made up of an array of circular stainless steel tubes with the dimensions and details specified in Reference 3. The key rack assembly dimensions and tolerances from the drawings are shown in Table 4.2. CASMO-4 calculations have demonstrated that this storage rack is significantly less limiting (reactive wise) than the square tube high density Boral storage racks. Hence, any BWR assembly that can be safely stored in the high density fuel storage racks can also be safely stored in the circular tube fuel storage rack.

4.3 *High Density Fuel Storage Rack*

The spent fuel storage rack dimensions and details are shown in Reference 3. The key rack assembly dimensions and tolerances from the drawings are shown in Table 4.3. The fuel pool storage cell with ATRIUM-10 fuel is modeled as shown in Figure 4.2. The spent fuel pool contains several sets of racks. Each rack consists of stainless steel boxes containing B_4C welded together at the corners to form the rack. Hence, alternative cells have no associated wall or B_4C. The pool contains racks in 13x15, 13x17, 13x19, and 15x17 fuel cell arrays. There are no B_4C absorber sheets on alternate outer edge cells of the rack, see Figure 4.3.

Table 4.1 ATRIUM™-10 Fuel Assembly Parameters

<u>Fuel Assembly</u>	
Fuel Rod Array	10x10
Fuel Rod Pitch, in.	0.510
Number of Fuel Rods Per Assembly	91
Water Channel	1
<u>Fuel Rods</u>	
Fuel Material	UO_2
Pellet Density, % of Theoretical	96.26*
Pellet Diameter, in.	0.3413
Pellet Void Volume, %	
Enriched UO_2	1.2†
Cladding Material	Zircaloy-2
Cladding OD, in.	0.3957
Cladding ID, in.	0.3480
<u>Internal Water Channel</u>	
Outside Dimension, in.	1.378
Inside Dimension, in.	1.321
Channel Material	Zircaloy-2 or Zircaloy-4
<u>Fuel Channel</u> (AREVA Advanced)‡	
Thick Corner thickness, in.	0.100
Thin Wall thickness, in	0.075
Inside Dimension, in.	5.278
Channel Material	Zircaloy-2 or Zircaloy-4

* Criticality safety analysis is valid for nominal pellet densities up to 96.26% TD.

† Depending on pellet L/D, the pellet void volume can vary. A nominal value of 1.2% was assumed for the criticality safety analysis. Variations of the void volume are not significant relative to impact on storage array criticality safety.

‡ The conclusions in this report are equally valid for fuel channels that may differ. Hence, conclusions remain valid for other fuel channel types, e.g., Standard 100 mil channels or similar design.

Table 4.2 Circular Tube Fuel Storage Rack Parameters

Parameter	Value
Rack outside diameter, in.	8.625 minimum
Rack wall thickness, in.	0.129 minimum
Material	Stainless steel (ss)
Rack cell pitch, in.	8.625 minimum

Table 4.3 High Density Fuel Storage Rack Parameters

Parameter	Value
B-10 areal density, g/cm^2	0.013 minimum
Inner rack box wall thickness, in.	0.030 ± 0.004
Box material	Stainless steel (ss)
Boral plate thickness, in.	0.076 ± 0.005
Plate material	B_4C clad in aluminum
Outer rack box wall thickness, in.	0.090 ± 0.008
Box Material	Stainless steel (ss)
Rack cell pitch, in.	6.563 ± 0.03
Rack cell outside dimension, in	6.653 ± 0.04

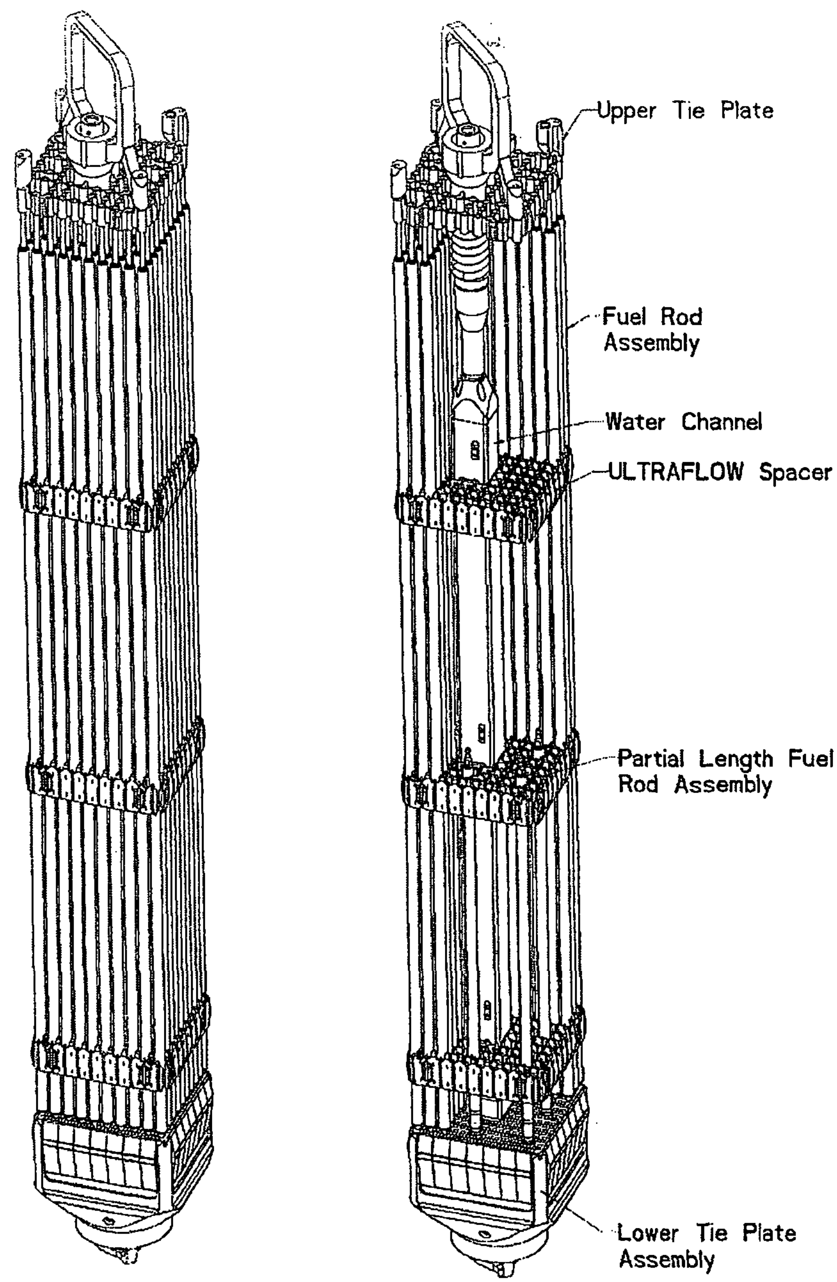


Figure 4.1 Representative ATRIUM-10 Fuel Assembly

(Assembly length and number of spacers has been reduced for pictorial clarity.)

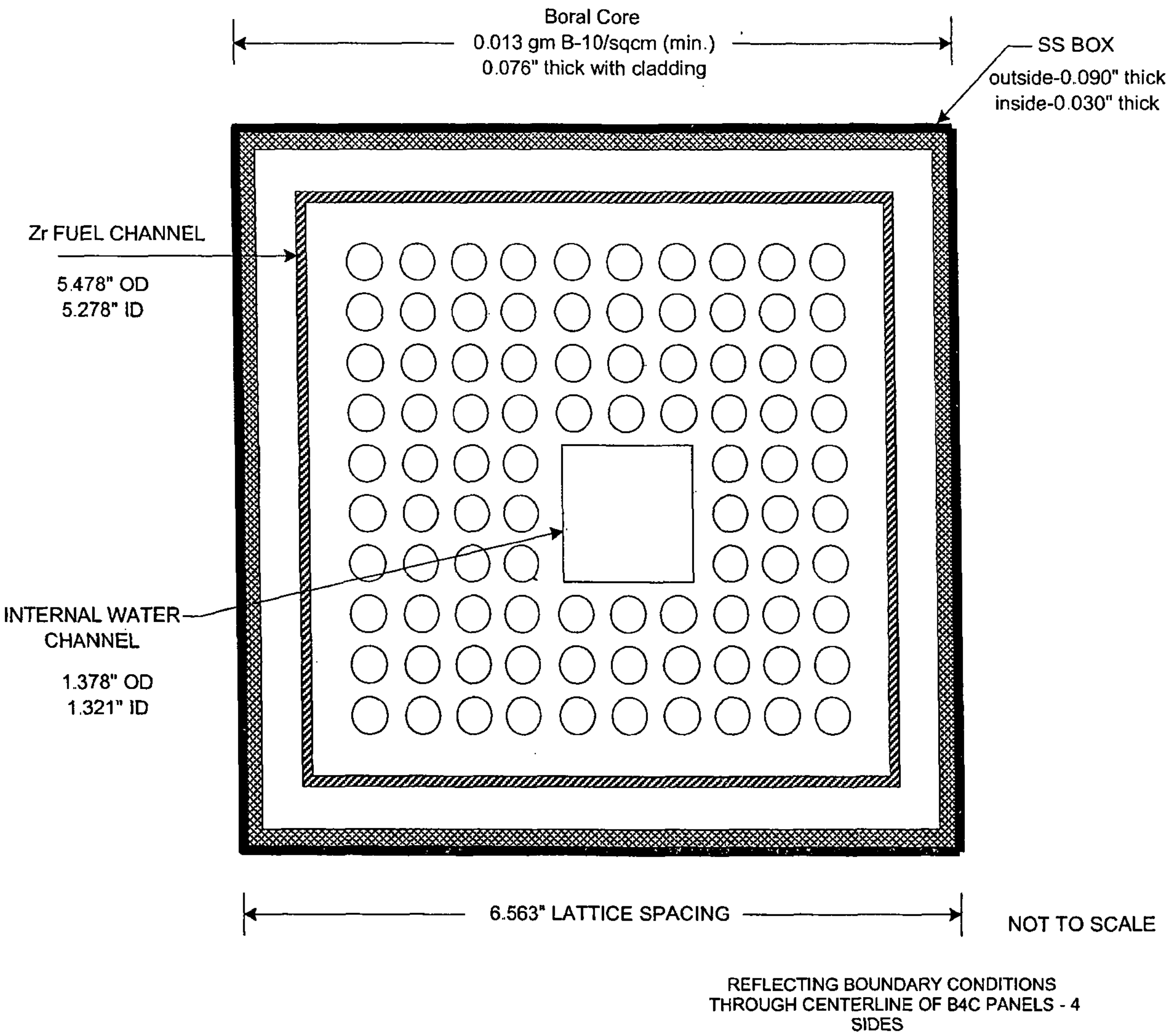
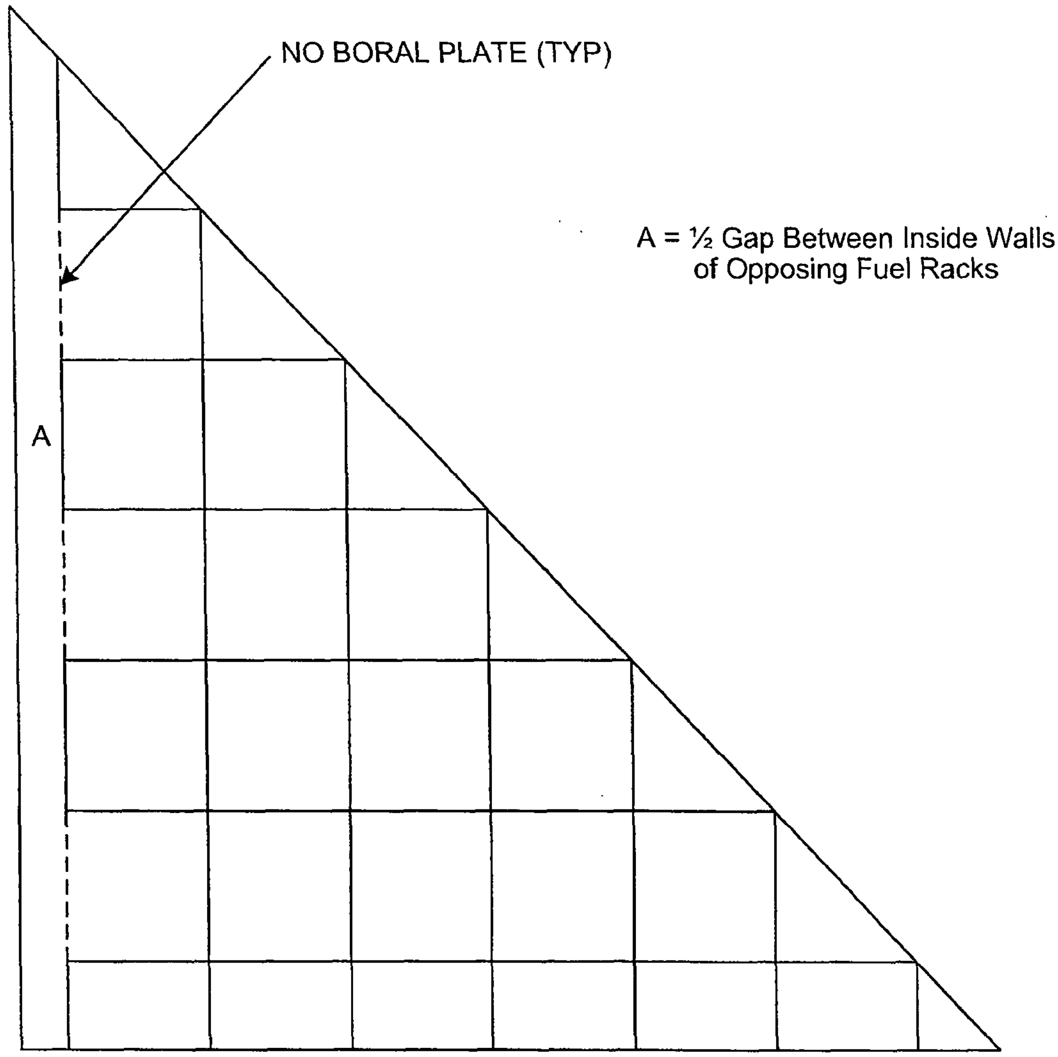


Figure 4.2 CASMO-4 Calculational Model of Storage Cell



Note 1/8 MODULE ARRAY (mirror reflection assumed on all boundaries)

Figure 4.3 KENO Calculational Model for Storage Array (Multiple Racks)

5.0 Calculation Methodology

The CASMO-4 bundle depletion code (Reference 10) is used to calculate k_∞ values for the ATRIUM-10 fuel assembly lattices as a function of exposure and void history for both in-core and in-rack geometries. CASMO-4 is a multigroup, two-dimensional transport theory code with an in-rack geometry option, where typical storage array geometries can be defined. The code has been benchmarked by Studsvik against cold critical data for both PWR and BWR fuel assemblies.

The spent fuel storage rack assembly calculations are performed with the KENO.Va Monte Carlo code, which is part of the SCALE 4.2 Modular Code System (Reference 11). Cross section data input to KENO.Va were taken from the 27 energy group data library and adjusted using the BONAMI and NITAWL codes to perform resonance corrections, using standard SCALE 4.2 methodology to account for resonance absorption in the uranium.

Both the KENO.Va and CASMO-4 computer codes are widely used throughout the nuclear industry for criticality safety and core physics calculations, respectively. AREVA has broad experience with both of these codes.

6.0 **Criticality Safety Analysis**

The criticality safety evaluation uses a bounding reference ATRIUM-10 lattice to demonstrate that the upper limit 95/95 k-eff for the Brunswick Nuclear Plant spent fuel pool can be met. These evaluations include the worst credible conditions and uncertainties as defined in the references documented in Section 3.0, the ATRIUM-10 fuel assembly design as defined in Section 4.1, and calculations with the CASMO-4 and KENO.Va codes.

6.1 ***Geometry Model***

The ATRIUM-10 fuel assembly parameters are given in Table 4.1. The key fuel pool storage rack parameters for the high density storage rack are given in Table 4.3. The KENO storage rack geometry model used for analysis is an infinite array of fuel storage boxes. This model bounds the actual rack configuration in the storage pool as described in Section 4.3 and Figure 4.3. All rack cells are assumed to contain an ATRIUM-10 fuel assembly. There are single Boral absorber sheets between all internal fuel assemblies in the rack assembly. There are no Boral sheets in every other cell location next to the fuel on the periphery of the rack assemblies, see Figure 4.3.

6.2 ***Storage Array Reactivity***

Single rack cell calculations are performed with the CASMO-4 code. The bounding reference ATRIUM-10 bundle has an enrichment of 4.50 wt% U-235 and eight gadolinia rods at 4.0 wt% Gd_2O_3. The CASMO-4 lattice depletion calculations are performed at hot operating, uncontrolled, [ ] void history conditions. The calculation results are for nominal fuel design parameters, as defined in Table 4.1 and assume the AREVA advanced fuel channel. Cold xenon-free restart calculations are performed as a function of exposure and void history to establish the highest in-rack lattice reactivity (k_∞). Calculations are performed for both top and bottom lattices at temperatures between 4° C and 100° C. The maximum calculated in-rack k_∞ of the top and bottom lattices are 0.880 and 0.876, respectively. This is based upon a 4°C water temperature, void histories of 40%, and lattice exposures between 11 and 12 GWd/MTU. The results of the CASMO-4 calculations are summarized in Table 6.1.

For the KENO rack calculations, an ATRIUM-10 reactivity equivalent beginning of life (REBOL) bottom lattice enrichment is selected that has a higher reactivity than the maximum reactivity of the bounding bundle (AT10-450B-8G4 where B refers to Bundle). The selected REBOL enrichment is 3.1 wt% U-235. The REBOL bundle has uniform enrichment in all rods, no natural blankets, and no gadolinia. The single assembly in-rack reactivity of the 3.1 wt% U-235 REBOL infinite lattice is 0.890 at 4°C as shown in Table 6.1. The REBOL infinite lattice reactivity is ~0.01 Δk_∞ higher than the bounding infinite lattice reactivity. This conservative assumption is used to account for any uncertainties associated with the REBOL approach.

For ease of modeling in KENO, the array calculations used the 100 mil channel. The array k-eff is about 4 mk higher with channels in place relative to the no channel case. There is no significant difference in array reactivity between the AREVA Advanced Channel and the 100 mil channel. Hence, subsequent KENO calculations used the 100 mil channel.

For the KENO rack array calculations, an infinite array of fuel storage cells was assumed. This model conservatively bounds the actual rack configuration currently in the pool. Assemblies in adjacent racks not separated by B_4C plates were considered in a separate model to assure that the infinite array model bounds the actual rack array with the water gaps between racks. All fuel locations in the rack assembly contain the 3.1 wt% U-235 REBOL fuel assembly. The KENO rack calculations take into account bundle lean next to the stainless steel box, minimum rack separation (top grid to top grid contact), and fuel orientation and positioning in the racks. The limiting conditions for the KENO rack calculations are shown in Table 6.2.

Except as specifically noted, the reactivity values presented in Tables 6.1 and 6.2 do not include adjustments for uncertainties or code biases. Section 6.5 presents the determination of the upper limit 95/95 reactivity for the storage rack assembly.

6.3 *Uncertainties*

Except as specifically noted, the reactivity result reported in Table 6.2 include the effects of bundle position and orientation in the rack and no additional uncertainty value is required for position or orientation. The uncertainties due to the fuel manufacturing process include variations in enrichment, UO_2 density, pellet diameter, clad thickness, pellet void volume, and gadolinia. The uncertainties due to storage rack tolerances include the stainless steel box dimensions. The ATRIUM-10 rack calculations are conservatively performed for a minimum B10 areal density and a minimum rack to rack spacing. The manufacturing tolerances and the calculated reactivity uncertainties for the ATRIUM-10 fuel are shown in Table 6.3. The overall reactivity manufacturing uncertainty value is [ ] Δk_{∞}.

6.4 *Accident Conditions*

In addition to the nominal storage cell arrangement, accident conditions are considered. The misplacement of a fuel assembly outside and adjacent to the fuel pool rack assembly was analyzed. The misplaced assembly is located in the larger water region next to the rack, where the side water reflector is greater than eighteen inches. Also considered was the situation where a single Boral plate in an interior position was missing. This was found to be the most reactive accident condition. The misplaced assembly gave a lower reactivity increase than the missing Boral plate.

The orientation of the bundles within the racks is not restricted even though the channel is not ¼ lattice symmetric. [

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For the case of a fuel assembly resting horizontally on top of the storage racks, the reactivity of the racks is not impacted because the dropped assembly is neutronically isolated from the fuel in the racks (greater than 12 inches of water between the assembly and the top of the active fuel zone of the fuel in the racks).

6.5 *Determination of Maximum Rack Assembly k-eff*

For the ATRIUM-10 fuel design with a REBOL bundle enrichment of 3.1 wt% U-235, the maximum KENO calculated in-rack reactivity from Table 6.2 is 0.893. This k-eff value is used with the following equation to determine the upper limit 95/95 reactivity:

$$k_{95/95} = k_c + b_b + A + U\sqrt{\sigma_c^2 + \sigma_b^2 + \sigma_t^2}$$

where:

k_c = in-rack reactivity from KENO.Va, (0.893, Table 6.2)

[

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U = 95/95 one-sided uncertainty multiplier (2.371)

σ_c = k-eff standard deviation from KENO.Va, (+0.001, Table 6.2)

[

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The standard deviations and tolerance uncertainties are included as the sum of the squares since they result from independent events. Solving for $k_{95/95}$ yields a 95/95 upper limit k-eff of 0.923 for the ATRIUM-10 fuel. The above determination of the upper limit 95/95 k-eff is consistent with the method documented in Reference 6 and allows one to state that at least 95% of the normal population is less than the 95/95 k-eff value calculated with a 95% confidence.

The results demonstrate the postulated configuration with the ATRIUM-10 REBOL assembly lattice with an enrichment of 3.1 wt% U-235 meets the NRC criticality safety acceptance criterion that array k-eff under worst credible conditions is $\leq$ 0.95. Since the REBOL infinite lattice has a higher reactivity than the bounding reference lattice as shown in Table 6.1, the bounding reference lattice also meets the $\leq$ 0.95 criterion.

6.6 *Interaction of BWR and PWR Spent Fuel Assemblies*

The Brunswick spent fuel pool also contains storage racks for PWR fuel assemblies. Interaction of the BWR and PWR storage racks is addressed in Section 7.4 of Attachment 2 of the Reference 2 Spent Fuel Storage Expansion License Amendment Request as follows: *"The minimum separation between the proposed and existing systems will be six inches. The proposed module and the existing storage system have individually been shown to have a neutron multiplication factor lower than the nuclear criticality safety criterion of 0.95. Each system has incorporated neutron absorber materials (stainless steel for the existing; boral and stainless steel for the proposed) in its design. The adjacent faces that the systems present to one another are either stainless steel or stainless steel and boral. In this configuration and with a separation of six inches of water, there is no significant neutron communication between systems. Calculations were made to determine the interaction between the faces of two high density modules, with partially unpoisoned storage locations directly opposite each other. These calculations (described in Section 7.3 and tabulated in Table 7-3) support the conclusion that neutron multiplication factor is insensitive to intermodule water gap."* The spacing study referred to above determined decreasing cell spacing from 2.967" to 1.244" increased k∞ by only 0.0087 (less than the calculation uncertainty), therefore neutron communication between storage systems incorporating stainless steel or boral poison material and spaced a minimum of 6" apart was concluded to be insignificant.

The ATRIUM-10 REBOL lattice enrichment of 3.1 wt% is comparable to the 3.0 wt% BWR enrichment analyzed in Reference 2, therefore these conclusions remain applicable.

6.7 *Arrays of Mixed BWR Fuel Types*

GE14 and ATRIUM-10 lattices have similar fuel rod pitch and water volumes; therefore, the neutron energy spectra associated with a mixed array of these assemblies will be similar. Hence, the most reactive single assembly type array bounds the mixed array condition.

Mixed arrays with Brunswick fuel assembly types older than GE14 are bounded by arrays of GE14 or ATRIUM-10 because the older fuel assembly types were designed at lower reactivity levels.

Table 6.1 Summary of CASMO-4 Maximum Reactivity Results for the ATRIUM™-10 Infinite Lattice

Bounding Reference Fuel Assembly Conditions

ATRIUM-10 lattice
4.5 wt% U-235 uniform enrichment
8 gadolinia rods with 4.0 wt% Gd_2O_3
AREVA 100 mil Channel*
No xenon in cold calculations
Top and bottom lattices evaluated

Limiting Conditions

Top Lattice
Exposure 12 GWd/MTU
40% void history

Calculated Reactivity

Condition	Maximum k_∞
In-Core, 20°C (68°F)	1.291 (top lattice)
In-Rack, 20°C (68°F)	0.879 (top lattice)
In-Rack, 4°C (39.2°F)	0.880 (top lattice)
In-Rack, 4°C (39.2°F)	0.876 (bottom lattice)

REBOL Lattice Conditions

ATRIUM-10 lattice
3.1 wt% U-235 uniform enrichment
No gadolinia
AREVA 100 mil Channel*
No xenon
BOL (zero exposure)

Calculated REBOL Reactivity

Condition	Maximum k_∞
In-Rack, 4°C	0.890

* Relative to array reactivity there is no significant difference between the 100 mil and AREVA Advanced Channel.

Table 6.2 Summary of KENO.Va Maximum In-Rack Reactivity for the ATRIUM™-10 Fuel

Fuel Assembly

ATRIUM-10 REBOL Lattice
3.1 wt% U-235 uniform enrichment
No gadolinia
No xenon
Zero exposure
Uniform thickness fuel channel (gives higher array k-eff)

Storage Array (Worst Case)

Infinite array of 13x13 racks (modeled as infinite storage cell)
Minimum rack separation (2" between racks)
Limiting fuel assembly orientation in storage cells (water box orientation)
4°C moderator and fuel temperatures

Maximum Rack Reactivity

Case	Description	k-eff
1	In-Rack 4°C (39.2°F) k_{eff}	0.893 $\pm$ 0.001
2	Maximum $k_{95/95}$ Reactivity (including uncertainties, biases, and worst accident)	0.923

Table 6.3 Manufacturing Reactivity Uncertainties

Quantity	Nominal Value	Tolerance	Δk_∞
Areal B-10 density	≥0.013 g/cm^2	Min value was used	--
Inner/outer SS Plate thickness	.036/0.090 in.	±0.004, ±0.008	0.00143*
Rack cell pitch	6.563 in.	6.452 in. was used†	--
Fuel enrichment	4.5 wt% U235	[	]
Fuel density	96.26% TD	[	]
Pellet diameter	0.3413 in.	[	]
Clad thickness	0.3957/0.3480 in.	[	]
Pellet void volume	1.2% TD	[	]
Gadolinia concentration	4.0 wt%	[	]
Statistical combination of uncertainties§			[ ]

* This value is equally valid for an inner wall thickness of 0.030 in.

† Conservatively represents the minimum rack pitch based on rack tolerances.

‡ This value is equally valid for a fuel density of 95.85% TD.

§ Square root of the sum of squares of all independent tolerance effects.

7.0 Conclusions

The Brunswick Nuclear Plant ATRIUM-10 fuel defined by a bounding reference lattice of ≤4.5 wt% U-235 with ≥8 gadolinia rods at ≥4.0 wt% Gd_2O_3 does not exceed an array k-eff of 0.95 in the Brunswick Nuclear Plant spent fuel storage pool with or without a fuel channel.

8.0 References

1. Letter, J. A. Jones (CP&L) to B. C. Rusche (NRC), "Brunswick Steam Electric Plant Unit Nos. 1 and 2 Docket Nos. 50-324 and 50-325 License Nos. DPR-62 and DPR-71 Request for Modification to License – Spent Fuel Storage Expansion," File NG-3514 (B), Serial NG-76-1281, Sept. 23, 1976.

2. Letter, E. E. Utley (CP&L) to T. A. Ippolito (NRC), "Brunswick Steam Electric Plant Unit Nos. 1 and 2 Docket Nos. 50-325 and 50-324 License Nos. DPR-71 and DPR-62 Request for License Amendment – Spent Fuel Storage Expansion," File NG-3514 (B), Serial NG-81-688, April 16, 1981.

3. ANP-2547(P) Revision 1, *Brunswick Unit 1 Cycle 17 Plant Parameters Document,* April 2007 (103-2547P-000).

4. NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,* Section 9.1.2 (Spent Fuel Storage), U.S. Nuclear Regulatory Commission, July 1981.

5. Spent Fuel Storage Facility Design Basis, Regulatory Guide 1.13, Proposed Revision 2, U.S. Nuclear Regulatory Commission, December 1981.

6. Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, ANSI/ANS American National Standard 57.2-1983, American Nuclear Society, October 1983.

7. Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors, ANSI/ANS American National Standard 8.17-1984, American Nuclear Society, January 1984.

8. Letter, Brian K. Grimes, Assistant Director for Engineering and Projects Division of Operating Reactors, U.S. Nuclear Regulatory Commission, to All Power Reactor Licensees, "OT Position for the Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

9. Letter, Laurence Kopp (Reactor Systems Branch, NRC) to Timothy Collins, Chief (Reactor Systems Branch-NRC), Subject: "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998.

10. EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,* Siemens Power Corporation, October 1999.

11. A Modular System for Performing Standardized Computer Analyses for Licensing Evaluation, SCALE 4.2, Oak Ridge National Laboratory, revised December 1993.

Appendix A Sample CASMO-4 Input

ATRIUM-10 fuel which does not conform to the bounding fuel lattice conditions in Table 2.1 (A10T-450L-8G4) can be analyzed for storage in the spent fuel pool racks by using the CASMO-4 sample input presented in Table A.1. (Note that the sample input assumes a uniform 100 mil fuel channel which has no impact on results since the conclusion is based on a difference between two cases.) A different version of CASMO-4 may be used as long as the base case in-rack k-inf (0.880) is re-established with the new code version for comparison. If the cold (4°C) in-rack reactivity of the new lattice is less than the base case value, ATRIUM-10 fuel can be safely stored in the Brunswick Nuclear Plant spent fuel storage rack.

Table A.1 Sample ATRIUM-10 CASMO-4 Input

```
TTL * A10T-450L-8G40 - .40 VB
TFU= 850.4
TMO= 560.9
VOI=40
FUE, 1, 10.42349/ 4.5000
FUE, 2, 10.27624/ 4.5000, 64016= 4.0000
BWR, 10,1.29540,13.40612,0.25400,0.80010,0.52578,1.2700
PDQ,'BND',1//92235, 92236, 92238, 94239, 94240, 94241, 94242, 95241
   54135, 62149, 93237, 94238, 64154, 64155, 64156, 64157, 64158
THE,0
FUM,0,2
PIN, 1,0.43345,0.44196,0.50254
PIN, 2, 1.67767,1.75006/'MOD','BOX'//-9
PIN, 3, 0.44196,0.50254/'COO','COO'
LPI
 1
 1 3
 1 1 1
 1 1 1 1
 1 3 1 1 2
 1 1 1 1 2 2
 1 1 1 1 2 2 2
 1 1 1 1 1 1 1 1
 1 3 1 1 1 3 1 1 3
 1 1 1 1 1 1 1 1 1 1
LFU
  1
  1  0
  1  2  1
  1  1  1  1
  1  0  1  1  0
  1  1  1  1  0  0
  1  2  1  1  0  0  0
  1  1  1  1  1  1  1  1
  1  0  1  2  1  0  1  2  0
  1  1  1  1  1  1  1  1  1  1
PDE,   59.2027, 'KWL'
DEP,0,0.1,0.5,1,1.5,2,2.5,3,3.5,4,4.5,5,5.5,6.0,6.5,7,7.5,8
    -10,-12.5,-15,-17.5,-20,-25.0
STA
TTL *+Brunswick High Density Storage Rack at 4 deg. C (Min. Water Gap)
RES,,0,.5,1,2,4,6,8,10,10.5,11.0,12.0,12.5,13.0,15,17.5,20,22.5,25
VOI,00
TMO, 277.1 TFU, 277.1 PDE,0
CNU,'FUE',54135,1.0E-14
BWR, 10,1.29540,13.40612,0.25400,0.994,0.994,1.2700
MI1 7.9/347=100.0
MI2 0.4692/5000=78.3  6000=21.7
FST  4*0.152/4*0.0965/8*'MI1'/8*'MI2'/
STA
END
```

Distribution

Controlled Distribution

Richland

O. C. Brown
R. E. Fowles
K. D. Hartley
C. D. Manning
T. E. Millsaps
J. L. Parker
P. D. Wimpy