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U. S. Nuclear Regulatory Commission
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LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: License Amendment Request Regarding Spent Fuel Storage Pool Criticality

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change revises Technical Specifications (TS) Section 4.3.1, "Criticality," to add a new requirement to use a blocking device in spent fuel storage rack cells that cannot maintain the effective neutron multiplication factor, K_{eff}, requirements specified in TS Section 4.3.1.1.a. In addition, the proposed change revises TS Section 4.3.3 to reflect that the Unit 2 spent fuel storage capacity is limited to no more than a combination of 4078 fuel assemblies and blocking devices.

The proposed change is necessary to resolve a non-conservative TS, in accordance with NRC Administrative Letter (AL) 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety." Specifically, as a result of Boraflex degradation in the LSCS Unit 2 spent fuel storage racks, EGC has determined that some of the storage rack cells are unusable, and additional cells will become unusable in the future. Therefore, the existing fuel storage criticality requirements contained in TS Section 4.3.1 are not sufficient to ensure that K_{eff} is less than or equal to 0.95 if fully flooded with unborated water, as required by TS Section 4.3.1.1.a. The proposed change to TS Section 4.3.1 will add an additional requirement to use a blocking device. Administrative controls are currently in place to prevent loading spent fuel in the storage rack cells that are unusable. In accordance with AL 98-10, EGC is requesting a license amendment to revise TS Sections 4.3.1 and 4.3.3 to address the non-conservative TS.

The proposed change to TS Section 4.3 is limited to Unit 2, since the Boraflex degradation issue is only applicable to Unit 2. Unit 1 fuel storage racks are designed with Boral neutron poison material.

On October 30, 2007, a pre-application meeting was held between the NRC and EGC. The purpose of the pre-application meeting was to provide an overview of the LSCS Unit 2 spent fuel pool storage and Boraflex degradation issue, summarize EGC's integrated approach to resolution, describe details of the 3-of-4 criticality analysis, and obtain NRC feedback with respect to the scope and level of detail of information needed to support a proposed license amendment request. Information requested by the NRC during the pre-application meeting is included in this submittal.

This request is subdivided as follows.

- Attachment 1 provides a description and evaluation of the proposed change.
- Attachment 2 provides a markup of the affected TS page.
- Attachment 3 provides a summary of the detailed criticality analysis performed by Holtec International in support of the proposed change.
- Attachment 4 provides an evaluation that demonstrates the ATRIUM-10 fuel assembly used in the Attachment 3 criticality analysis conservatively bounds the current inventory of ATRIUM-10 fuel assemblies in the LSCS Units 1 and 2 reactors and spent fuel pools.

The proposed change has been reviewed by the LSCS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed change by December 15, 2008. Once approved, the amendment will be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of December 2007.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Evaluation of Proposed Change
2. Markup of Proposed Technical Specifications Page
3. Holtec International Report No. HI-2073758, "Licensing Report for LaSalle 3 of 4 Storage with Loss of Boraflex," Revision 2
4. AREVA NP Inc. Report No. ANP-2684, "LaSalle Unit 2 Nuclear Power Station Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™-10 Fuel in a 2x2-1 Configuration without Boraflex," Revision 0

ATTACHMENT 1
Evaluation of Proposed Change

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change revises Technical Specifications (TS) Section 4.3.1, "Criticality," to add a new requirement to use a blocking device in spent fuel storage rack cells that cannot maintain the effective neutron multiplication factor, K_{eff}, requirements specified in TS Section 4.3.1.1.a. In addition, the proposed change revises TS Section 4.3.3 to reflect that the Unit 2 spent fuel storage capacity is limited to no more than a combination of 4078 fuel assemblies and blocking devices.

The proposed change is necessary to resolve a non-conservative TS, in accordance with NRC Administrative Letter (AL) 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety" (i.e., Reference 1). Specifically, as a result of Boraflex degradation in the LSCS Unit 2 spent fuel storage racks, EGC has determined that some of the storage rack cells are unusable, and additional cells will become unusable in the future. Therefore, the existing fuel storage criticality requirements contained in TS Section 4.3.1 are not sufficient to ensure that K_{eff} is less than or equal to 0.95 if fully flooded with unborated water, as required by TS Section 4.3.1.1.a. The proposed change to TS Section 4.3.1 will add an additional requirement to use a blocking device. Administrative controls are currently in place to prevent loading spent fuel in the Unit 2 storage rack cells that are unusable. In accordance with AL 98-10, EGC is requesting a license amendment to revise TS Sections 4.3.1 and 4.3.3 to address the non-conservative TS.

The proposed change to TS Section 4.3 is limited to Unit 2, since the Boraflex degradation issue is only applicable to Unit 2 fuel storage racks. Unit 1 fuel storage racks are designed with Boral neutron poison material. The Unit 1 fuel storage racks remain capable of meeting the criticality requirements of TS Section 4.3.1 when fully loaded with fuel.

2.0 PROPOSED CHANGE

The LSCS TS requirements related to spent fuel storage are contained in TS Section 4.3, "Fuel Storage." TS Section 4.3.1, "Criticality," currently identifies requirements related to the design of the spent fuel storage racks. Specifically, Section 4.3.1.1.a requires K_{eff} to be less than or equal to 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the Updated Final Safety Analysis Report (UFSAR). Section 4.3.1.1.b requires a nominal 6.26 inch center to center distance between fuel assemblies placed in the storage racks.

The proposed change adds a new requirement, 4.3.1.1.c, which states:

> c. For Unit 2 only, a blocking device shall be installed in spent fuel storage rack cells that cannot maintain the requirements of 4.3.1.1.a.

TS Section 4.3.3, "Capacity," currently identifies limitations on the spent fuel storage pool storage capacity for both units. The existing TS limits the Unit 2 storage capacity to no more than 4078 fuel assemblies. The proposed change revises the limit for Unit 2 to reflect that the storage capacity is limited to no more than a combination of 4078 fuel assemblies and blocking devices. The revised TS Section 4.3.3 states:

> The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3986 fuel assemblies for Unit 1 and a combination of 4078 fuel assemblies and blocking devices for Unit 2.

3.0 BACKGROUND

LSCS has two spent fuel storage pools, one for each unit, that provide for storage of irradiated fuel in a safe manner. The two spent fuel pools (SFPs) are connected by a double-gated transfer canal. The SFP facilities are designed to accept irradiated fuel from both the Unit 1 and Unit 2 reactor cores.

The Unit 1 SFP contains high density racks consisting of 21 individual racks that have capacity for 3986 fuel assemblies and 43 special storage cells. The fuel storage cells consist of 3982 normal fuel storage cells and four defective fuel storage cells. The special storage cells consist of 39 control rod storage cells (i.e., one rack of 18 and one rack of 21), and four control rod guide tube storage cells. The Unit 1 high density racks contain a 0.079 inch thick sheet of Boral neutron poison material with a B-10 loading of 0.022 grams per square centimeter physically captured between the side walls of each box and sheathing welded to the sides of the box.

The Unit 2 SFP contains high density racks consisting of 20 individual racks that have capacity for 4078 fuel assemblies and 38 special storage cells. The fuel storage cells consist of 4073 normal fuel storage cells and five defective fuel storage cells. The special storage cells consist of 35 control rod storage cells (i.e., one rack of 18 and one rack of 17), and three control rod guide tube storage cells. The Unit 2 high density racks contain a nominal 0.075 inch thick sheet of Boraflex neutron poison material with a nominal B-10 loading of 0.0238 grams per square centimeter physically captured between the side walls of all adjacent boxes. To provide space for the poison sheet between boxes, a double row of matching flat round raised areas are coined in the side walls of all boxes. The raised dimension of these locally formed areas on each box wall is half the thickness of the poison sheet.

The spent fuel racks are designed to maintain the stored spent fuel in a space geometry that precludes the possibility of criticality. The racks maintain this subcritical array when subjected to maximum earthquake conditions, dropped fuel assembly accident conditions, and any uplift forces generated by the fuel handling equipment.

The fully loaded array of stored fuel assemblies is calculated to maintain K_{eff} less than or equal to 0.95 assuming the pool is filled with unborated water at 39.2°F, under both normal and abnormal conditions. Analyses have been performed for each type of fuel stored in the Unit 2 SFP to assure compliance with the K_{eff} requirement.

NRC Generic Letter 96-04 (i.e., Reference 2) discusses that when Boraflex is subjected to gamma radiation in a spent fuel pool environment, the silicon polymer matrix becomes degraded and silica filler and boron carbide are released. Due to potential Unit 2 spent fuel storage rack Boraflex degradation, a comprehensive Boraflex monitoring program has been implemented at LSCS. The Boraflex monitoring program includes the following elements:

- Periodic offsite testing of part-length Boraflex surveillance coupons,
- Periodic onsite inspection of full-length Boraflex surveillance coupons,
- Periodic neutron blackness testing of a sampling of SFP rack cell walls, and
- Use of the Electric Power Research Institute (EPRI) RACKLIFE computer code to model Boraflex degradation.

Results from the Boraflex monitoring program indicate that some of the Unit 2 storage rack cells are currently unusable, since the cells have degraded to less than an acceptable threshold for Boron areal density. The acceptance criterion for Boron areal density degradation is 57.5%. Applying a 10% uncertainty to individual panel degradation, and a 20% uncertainty to average cell degradation, has resulted in three cells becoming unusable. In addition, based on the most recent RACKLIFE projection performed in October 2007, approximately 200 additional cells will become unusable by July 1, 2008.

Projections currently indicate that with the continued Boraflex degradation expected to occur over the next several years, additional cells will become unusable and full core discharge capability will be lost in 2010. The loss of full core discharge capability will impact both Units 1 and 2, since the two SFPs are connected and the projection for when full core discharge capability will be lost assumes that use of the Unit 1 SFP has been maximized.

As Boraflex degradation continues, EGC plans to implement administrative controls at LSCS to remove fuel from unusable cells and to prevent loading fuel into Unit 2 storage rack cells determined to be unusable. Specifically, when a cell is projected to become unusable, fuel move sheets are prepared by qualified reactor engineers, and independently reviewed by qualified reactor engineers, to evacuate fuel from the unusable cells and to install a blocking device in each cell location. This blocking device typically consists of a single blade guide; however, double blade guides or fuel channels may also be used. The ShuffleWorks database is also updated to classify cells as "Unusable Locations" in the Unit 2 SFP, which prevents the move sheet builder software from allowing fuel to be placed in these locations. In addition, site procedures that govern fuel handling in the SFP are revised to list the unusable locations due to Boraflex degradation.

In order to recover a portion of the cells that are unusable, the requested license amendment proposes a 3-of-4 spent fuel loading scheme that will be implemented in the unusable locations to ensure that the requirement to maintain K_{eff} less than or equal to 0.95, if fully flooded with unborated water, is met. A 3-of-4 criticality analysis has been prepared to support this loading scheme. The analysis demonstrates that K_{eff} remains less than or equal to 0.95 for the normal and abnormal cases evaluated, with no credit for the Boraflex neutron poison material. Implementation of this alternative loading scheme will allow spent fuel to be stored in up to 75% of the unusable locations. In addition, the Boraflex panels will remain in place providing additional, albeit diminished, neutron absorption capability that is not credited in the 3-of-4 criticality analysis.

4.0 TECHNICAL ANALYSIS

A criticality analysis has been performed to support the storage of spent fuel in the LSCS Unit 2 SFP in a 3-of-4 configuration with no credit for Boraflex in the racks. The analysis demonstrates that the effective neutron multiplication factor, K_{eff}, is less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity and the SFP flooded with unborated water at a temperature corresponding to the highest reactivity, with no credit for Boraflex. The maximum calculated reactivities included a margin for uncertainty in reactivity calculations, including manufacturing tolerances, and were calculated with a 95% probability at a 95% confidence level. In addition, reactivity effects of abnormal and accident conditions have also been evaluated to assure that under all credible abnormal and accident conditions, K_{eff} will not exceed 0.95. The 3-of-4 criticality analysis is provided in Attachment 3.

The key differences between the 3-of-4 criticality analysis and the current 4-of-4 analysis are: (1) the 3-of-4 analysis uses one empty cell with a blocking device in each 2-by-2 array, and (2) the 3-of-4 analysis does not credit any Boraflex in the racks. The analysis in Attachment 3 outlines the methodology and key assumptions used. The analysis was performed using an ATRIUM-10 fuel assembly as the principal design basis for the spent fuel storage racks, containing uranium dioxide fuel rods clad in Zircaloy, with a planar uniform enrichment of 2.45 wt% U-235, no burnup, and no Gadolinium burnable poison. This fuel bundle bounds the peak reactivity of every fuel assembly at LSCS, as discussed below.

Attachment 4 provides an evaluation that establishes criticality design limits for ATRIUM-10 fuel. These limits are combinations of U-235 enrichment and Gadolinium burnable poison that result in acceptable bundle designs. EGC has confirmed that these limits conservatively bound the current inventory of ATRIUM-10 fuel in the LSCS Units 1 and 2 reactors and SFPs.

The ATRIUM-10 fuel assembly in the Attachment 3 criticality analysis also bounds legacy fuel types used at LSCS prior to ATRIUM-10. The limiting lattice at LSCS, with respect to margin to spent fuel pool criticality, is currently an ATRIUM-10 lattice from Unit 1 Cycle 13. EGC has evaluated this lattice and determined that it is bounded by the 2.45 wt% U-235 uniform enriched ATRIUM-10 no Gadolinium lattice modeled in the criticality analysis. A summary of the limiting fuel lattices for the different fuel types stored in both the Unit 1 and Unit 2 SFPs is shown below.

Fuel Type	In-Core K-inf of Limiting Lattice
Legacy GE (8x8) Fuel	1.2421
ATRIUM-9B	1.2398
GE14	1.2045
ATRIUM-10	1.2764
ATRIUM-10 2.45 wt% U-235	1.2845

Each result presented above is based upon a 20 °C, in-core, peak reactivity exposure, peak void history, CASMO-3 or 4 analysis. The CASMO-4 analysis for the limiting ATRIUM-10 lattice, which produced the 1.2764 K-inf, clearly bounds the reactivity of the historic fuel in both SFPs, even considering small differences between a CASMO-3 versus a CASMO-4 result. The

ATRIUM-10 2.45 wt% U-235 uniform enrichment lattice has been shown to bound this limiting ATRIUM-10 lattice in an in-rack 3-of-4 geometry in Attachment 4. The margin between the ATRIUM-10 in-core peak reactivity and the historic fuel type peak reactivities is sufficient to ensure that the 2.45 wt% U-235 enriched ATRIUM-10 lattice will bound these fuel types in the in-rack 3-of-4 geometry as well.

Interfaces Between Areas of 3-of-4 and 4-of-4 Storage

The 3-of-4 criticality analysis assumes that all fuel storage cells exhibit an unacceptable level of Boraflex degradation; therefore, no credit is taken for the Boraflex. However, in reality there are areas in the Unit 2 SFP where the Boraflex has not degraded beyond acceptable levels. These areas will continue to be used to store spent fuel in a full 4-of-4 array. For the interfaces between areas of 3-of-4 storage and 4-of-4 storage, the following controls will be implemented to meet the proposed TS requirements to ensure the supporting analyses remain valid.

- Each cluster of four storage cells (i.e., 2-by-2) must meet either the criteria for 4-of-4 storage or the criteria for 3-of-4 storage.

- In each cluster of four storage cells (i.e., 2-by-2), if one storage cell is considered unusable (i.e., one or more of the four surrounding Boraflex panels is degraded beyond acceptable levels), then one of the four cells must contain a blocking device.

These operational controls will ensure that storage of spent fuel implements the proposed TS requirements while ensuring the supporting analyses remain valid.

Administrative and Physical Controls

The unusable areas of the storage racks will be controlled to ensure assumptions of the 3-of-4 criticality analysis are met. Specifically, EGC will implement robust administrative and physical controls in order to meet the assumptions of the 3-of-4 criticality analysis and to ensure that a fuel assembly is not loaded into a location required to be empty.

The administrative controls that will be implemented are similar to the controls currently in place to support movement and storage of spent fuel in the SFP. Site reactor engineers are responsible for identifying the correct locations for all fuel assemblies, and qualified fuel handlers are responsible for moving fuel, under the supervision of a qualified fuel handling supervisor. All fuel moves are preplanned, and planned moves are documented on move sheets before the fuel is moved. The move sheets are prepared by qualified reactor engineers and independently reviewed by qualified reactor engineers. The approved move sheets are then provided to the fuel handling crew. The crew moves the fuel in accordance with the move sheets. Each move is signed off by the crew prior to the next move. In addition, each move is verified by the fuel handler, a second fuel handler, and the fuel handling supervisor.

The 3-of-4 loading scheme does not require a more complex methodology to characterize fuel assemblies or identify the correct storage locations. The administrative process for controlling fuel movement provides several barriers to prevent mislocation of a fuel assembly.

Overall, implementation of the 3-of-4 loading scheme will result in fewer fuel moves than with the current configuration. Currently, when storage rack cells become unusable, EGC's administrative controls require removing all four fuel assemblies in each 2-by-2 array of the locations degraded beyond acceptable levels. With the proposed change, only one fuel assembly in each 2-by-2 array of the degraded locations will be removed.

In addition to the administrative controls discussed above, EGC intends to implement an additional physical control to prevent misloading of a fuel assembly into a location assumed empty in the 3-of-4 criticality analysis. The physical control consists of a blocking device that will be placed into an unusable cell to enforce the assumed loading scheme in the criticality analysis. EGC intends to implement controls for movement of a blocking device that are similar to the controls that govern fuel movement.

In the criticality analysis, the blocking device is an alloy 1100 double thickness schedule 10 aluminum pipe, with a 5 inch nominal diameter. Since this material has a lower neutron capture cross section than water, the criticality analysis conservatively models the pipe at twice its normal thickness. The design of the device will ensure visibility from the refuel bridge, and will ensure flow through the cell is not adversely impacted. Specifically, the design will ensure that the pressure drop of the device is less than the pressure drop through a fuel assembly when evaluated under natural circulation of water. The actual blocking device used (e.g., fuel channel, blade guide, etc.) may be different than the device modeled in the criticality analysis, as long as supporting analyses of the selected blocking device demonstrate that these design requirements are met and the device is conservative with respect to that modeled in the criticality analysis.

Use of the blocking device does not affect the fuel handling accident in the SFP. The fuel handling accident in the SFP involves dropped fuel assemblies, where one assembly falls onto another assembly, or an assembly falls onto the top of the spent fuel storage racks. The dose consequences are limited by the number of rods that fail, and the number of rods that fail is limited by the energy of the collision between the dropped assembly and the other assembly or structure that is hit by the dropped assembly. The design of the blocking device will ensure that the total number of fuel rods that fail would be less than the current design basis if either the device is dropped onto a fuel assembly, or if a fuel assembly is dropped onto a blocking device, since the weight of the blocking device is significantly less than the weight of a fuel assembly. In addition, the use of blocking devices does not affect the isotopic inventory of the affected fuel assemblies involved in the postulated fuel handling accident.

Use of the blocking device provides a robust physical control to prevent mislocation of a fuel assembly. In addition, the blocking device does not impact the existing fuel handling accident analysis in the SFP.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility

Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change revises Technical Specifications (TS) Section 4.3.1, "Criticality," to add a new requirement to use a blocking device in spent fuel storage rack cells that cannot maintain the K_{eff} requirements specified in TS Section 4.3.1.1.a. In addition, the proposed change revises TS Section 4.3.3 to reflect that the Unit 2 spent fuel storage capacity is limited to no more than a combination of 4078 fuel assemblies and blocking devices.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change adds an additional requirement to the TS to ensure that the effective neutron multiplication factor, K_{eff}, is less than or equal to 0.95, if fully flooded with unborated water. The additional requirement is to insert a blocking device into unusable storage rack cell locations. Since the proposed change pertains only to the spent fuel pool (SFP), only those accidents that are related to movement and storage of fuel assemblies in the SFP could be potentially affected by the proposed change.

   The probability that a misplaced fuel assembly would result in an inadvertent criticality is unchanged since the process and procedural controls governing fuel movement in the SFP will not be changed. The current criticality analysis for the LSCS Unit 2 SFP credits the neutron absorbing properties of the Boraflex neutron poison material in the spent fuel storage racks. The current analysis demonstrates: (1) adequate margin to criticality for all spent fuel storage cells, (2) adequate margin for fuel assemblies inadvertently placed into locations adjacent to the spent fuel racks, and (3) adequate margin for assemblies accidentally dropped onto the spent fuel racks. The dose consequences of the most limiting drop of a fuel assembly in the spent fuel pool is limited by the number of the fuel rods damaged and other engineered features unaffected by the proposed change, including the fuel design, fuel decay time, water level in the spent fuel pool, water temperature of the spent fuel pool, and the engineering features of the Reactor Building Ventilation System.

The revised analysis does not result in a significant increase in the probability of an accident previously analyzed. The revised analysis takes no credit for the Boraflex material. The use of a blocking device prevents an inadvertent action to insert a spent fuel assembly, and prevents an assembly that is accidentally dropped to penetrate into the empty spent fuel cell. In addition to this blocking device, administrative controls will be implemented to prevent insertion of a bundle into a cell that is blocked. The probability that a fuel assembly would be inadvertently placed into a location adjacent to the racks is unchanged, and the probability that a fuel assembly would be dropped is unchanged by the revised analysis. These events involve failures of administrative controls, human performance, and equipment failures that are unaffected by the presence or absence of Boraflex and the blocking devices.

The revised analysis does not result in a significant increase in the consequence of an accident previously analyzed. The revised analysis demonstrates adequate margin to criticality for unblocked cells in the LSCS Unit 2 SFP, adequate margin for assemblies inadvertently placed into locations adjacent to the spent fuel racks, and adequate margin for assemblies accidentally dropped onto the spent fuel racks. Placing a spent fuel assembly into a location containing a blocking device is not a credible event since there are diverse and redundant administrative and physical barriers to prevent that.

The revised analysis does not affect the consequences of a dropped fuel assembly. The consequences of dropping a fuel assembly onto any other fuel assembly or other structure, other than a blocking device, are unaffected by the change. The consequences of dropping a fuel assembly onto a blocking device are bounded by the event of dropping an assembly onto another assembly, both for criticality and for radiological consequences. For criticality, the blocking device prevents the dropped assembly from entering the blocked cell. For radiological consequences, the number of rods damaged when a fuel assembly is accidentally dropped onto a blocking device is bounded the by the number of rods damaged by an assembly dropped onto another assembly. The change does not affect the effectiveness of the other engineered design features to limit the offsite dose consequences of the limiting fuel assembly drop accident.

The proposed change to clarify that the capacity of the Unit 2 SFP is limited to no more than a combination of 4078 fuel assemblies and blocking devices does not affect the probability or consequences of an accident previously analyzed because no physical modifications to the storage racks are proposed. The proposed change will reduce the number of allowable fuel assembly storage locations.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Onsite storage of spent fuel assemblies in the SFP is a normal activity for which LSCS has been designed and licensed. As part of assuring that this normal activity can be performed without endangering public health and safety, the ability to safely accommodate different possible accidents in the SFP, such as dropping a fuel assembly or misloading a fuel assembly, have been analyzed. The proposed fuel storage configuration does not change the methods of fuel movement or fuel storage. No structural or mechanical change to the racks or fuel handling equipment is being proposed. The proposed change allows for partial use of storage rack locations that have been determined unusable based on the existing criticality analysis.

   The blocking devices are passive devices. These devices, when inside a spent fuel storage rack cell, perform the same function of a spent fuel assembly in that cell. These devices do not add any limiting structural loads or affect the removal of decay heat from the other assemblies. The devices are resistant to corrosion and will maintain their structural integrity over the life of the plant. These devices are not under any structural load during normal operations. They are only challenged by an accidental fuel assembly drop. The existing fuel handling accident, which assumes the drop of a fuel bundle, bounds the drop of a blocking device.

   This change does not create the possibility of a misloaded assembly into a blocked cell. Placing a spent fuel assembly into a location containing a blocking device is not a credible event since there are diverse and redundant administrative and physical barriers to prevent that.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   LSCS TS 4.3.1.1 requires the spent fuel storage racks to maintain the effective neutron multiplication factor, K_{eff}, less than or equal to 0.95 when fully flooded with unborated water, which includes an allowance for uncertainties. Therefore, for criticality, the required safety margin is 5% including a conservative margin to account for engineering uncertainties.

The proposed change adds a requirement to use a blocking device to ensure that K_{eff} continues to be less than or equal to 0.95; thus, the required safety margin of 5% is preserved. The proposed change also clarifies that the capacity of the Unit 2 SFP is limited to no more than a combination of 4078 fuel assemblies and blocking devices. This clarification does not impact the required safety margin of 5%.

The current analysis assumes an infinite array of fuel with all fuel at the peak reactivity (i.e., the highest combination of initial enrichment, gadolinium, and fuel burnup that maximizes the reactivity of the fuel). The revised analysis demonstrates the same margin to criticality of 5%, including a conservative margin to account for engineering uncertainties, is maintained assuming an infinite array of fuel with all fuel at the peak reactivity. In addition, the margin of safety for radiological consequences of a dropped fuel assembly are unchanged because the event involving a dropped fuel assembly onto a blocking device is bounded by the consequences of a dropped fuel assembly onto another fuel assembly.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

General Design Criterion (GDC) 61, "Fuel storage and handling and radioactivity control," specifies, in part, that fuel storage systems shall be designed with residual heat removal capability having reliability and testability that reflects the importance of safety of decay heat removal, and with the capability to prevent significant reduction in fuel storage coolant inventory under accident conditions. The evaluation of LSCS's conformance with GDC 61 is discussed in Section 3.1.2.6.2 of the LSCS UFSAR. The proposed change does not affect the conclusions of UFSAR Section 3.1.2.6.2 since no physical modifications to the fuel storage systems are proposed. The proposed change only affects the SFP criticality analysis that defines acceptable fuel storage patterns, and implements a physical blocking device that meets the same design requirements as a fuel assembly.

GDC 62, "Prevention of criticality in fuel storage and handling," states that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. In SRP Section 9.1.2, the NRC has established a 5% subcriticality margin (i.e., K_{eff} less than or equal to 0.95) for nuclear power plant operators to comply with GDC 62. The evaluation of LSCS's conformance with GDC 62 is discussed in Section 3.1.2.6.3 of the LSCS UFSAR. The 3-of-4 criticality analysis provided in Attachment 3, performed in accordance with SRP guidance, demonstrates that K_{eff} will remain less than or equal to 0.95 with no credit for the Boraflex neutron poison material present in the Unit 2 spent fuel storage racks.

10 CFR 50.68, "Criticality accident requirements," paragraph (b)(4) requires that, if no credit for soluble boron is taken, the K_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. The 3-of-4 criticality analysis provided in Attachment 3 demonstrates that this requirement is met.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

EGC has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation." However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," dated December 29, 1998

2. NRC Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," dated June 26, 1996

ATTACHMENT 2
Markup of Proposed Technical Specifications Page

LaSalle County Station, Units 1 and 2

Facility Operating License Nos. NPF-11 and NPF-18

REVISED TECHNICAL SPECIFICATIONS PAGES

4.0-2

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

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

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the UFSAR; and

b. A nominal 6.26 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 819 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3986 fuel assemblies for Unit 1 and ~~4078 fuel assemblies for Unit 2.~~

c. For Unit 2 only, a blocking device shall be installed in spent fuel storage rack cells that cannot maintain the requirements of 4.3.1.1.a.

a combination of 4078 fuel assemblies and blocking devices for Unit 2.

ATTACHMENT 3

Holtec International Report No. HI-2073758, "Licensing Report for LaSalle 3 of 4 Storage with Loss of Boraflex," Revision 2

HOLTEC
INTERNATIONAL

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LICENSING REPORT FOR LASALLE 3 OF 4 STORAGE WITH LOSS OF BORAFLEX

FOR

EXELON

Holtec Report No: HI-2073758

Holtec Project No: 1647

Report Class : SAFETY RELATED

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DOCUMENT CATEGORIZATION

In accordance with the Holtec Quality Assurance Manual and associated Holtec Quality Procedures (HQPs), this document is categorized as a:

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- ☑ Technical Report (Per HQP 3.2) (Such as a Licensing Report)
- ☐ Design Criterion Document (Per HQP 3.4)
- ☐ Design Specification (Per HQP 3.4)
- ☐ Other (Specify):

DOCUMENT FORMATTING

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3. Revisions to this document may be made by adding supplements to the document and replacing the "Table of Contents", this page and the "Revision Log".

Summary of Revisions

Revision 0: Original issue

Revision 1: This revision incorporates comments by Exelon. Proprietary designation removed from footer.

Revision 2: This revision incorporates additional comments by Exelon. Proprietary designation removed from Appendix A.

Table of Contents

1.0 INTRODUCTION AND SUMMARY

This report documents the criticality safety evaluation for the storage of BWR spent fuel in the LaSalle Unit 2 spent fuel pool for storage in a 3 out of 4 configuration with no credit for residual Boraflex in the racks.

The objective of this analysis is to ensure that the effective neutron multiplication factor (k_{eff}) is less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity and the pool flooded with unborated water at a temperature corresponding to the highest reactivity [7]. The maximum calculated reactivities include a margin for uncertainty in reactivity calculations, including manufacturing tolerances, and are calculated with a 95% probability at a 95% confidence level [6]. Reactivity effects of abnormal and accident conditions have also been evaluated to assure that under all credible abnormal and accident conditions, the reactivity will not exceed the regulatory limit of 0.95.

The fuel assembly used as the principal design basis for the racks is an ATRIUM[1]-10 (10x10) fuel assembly, containing UO_2 fuel rods clad in Zircaloy, and using a planar uniform enrichment of 2.45 wt% ^{235}U. This assembly has been determined to bound all past and present fuel assemblies of any type in both LaSalle Unit 1 and Unit 2. The effects of calculational and manufacturing tolerances were evaluated and added in determining the maximum k_{eff} in the storage rack. The following acceptance criteria is defined for acceptable storage in the LaSalle Unit 2 spent fuel pool in a 3 of 4 configuration:

1. A fuel assembly acceptable for storage in the LaSalle Unit 2 spent fuel pool must have a reactivity in the storage racks less than an ATRIUM-10 fresh fuel assembly with a maximum planar uniform enrichment of 2.45 wt% ^{235}U.

This criterion is sufficient to determine the acceptability of fuel for safe storage in the spent fuel racks. Figure 7.1 presents an optimal configuration of the LaSalle Unit 2 spent fuel pool.

The design basis calculations supporting the criticality safety of the LaSalle Unit 2 fuel storage racks are summarized in Table 7.1

Abnormal and accident conditions were also evaluated. None of the abnormal or accident conditions that have been identified as credible will result in exceeding the limiting reactivity (k_{eff} of 0.95). The double contingency principle of ANSI 16.1-1975 (and the USNRC letter of April 1978) specifies that it shall require at least two unlikely independent and concurrent events to produce a criticality accident. This principle precludes consideration of the simultaneous occurrence of multiple accident conditions.

2.0 METHODOLOGY

The analytical methodology used in this report consists primarily of using two computer codes to perform the calculations, CASMO-4 [1-4] and MCNP-4A [5]. CASMO-4 was used to calculate

[1] ATRIUM is a trademark of AREVA NP.

the reactivity effect of manufacturing tolerances and temperature variation. MCNP-4A was used to calculate the reactivity of the fuel in the racks and to determine the reactivity effect of eccentric fuel positioning and orientation of fuel within the rack.

The maximum k_{eff} is determined from the MCNP-4A calculated k_{eff}, the calculational bias, the temperature bias, and the applicable uncertainties and tolerances (bias uncertainty, calculational uncertainty, rack tolerances, fuel tolerances) using the following formula:

$$\text{Max } k_{eff} = \text{Calculated } k_{eff} + \text{biases} + [\Sigma_i (\text{Uncertainty}_i)^2]^{1/2}$$

In the geometric models used for the calculations, each fuel rod and its cladding were described explicitly and reflecting or periodic boundary conditions were used in the radial direction which has the effect of creating an infinite radial array of storage cells.

2.1 Code Validation

As stated, CASMO-4 was used for criticality calculations of tolerance and temperature effects. As proof of its acceptability in this application, CASMO-4 has been verified [3,4] against Monte Carlo calculations and critical experiments.

Benchmarking of MCNP-4A against critical experiments has been performed at Holtec. The results of the benchmark calculations, presented in Appendix A, indicate a bias of 0.0009 ± 0.0011 for MCNP-4A over a wide range of compositions and geometries, evaluated at the 95% probability, 95% confidence level [6].

3.0 ACCEPTANCE CRITERIA

The high-density spent fuel storage racks for LaSalle Unit 2 are designed to assure that the neutron multiplication factor (k_{eff}) is equal or less than 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity and the pool flooded with unborated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations and in manufacturing tolerances, statistically combined, giving assurance that the true k_{eff} will be equal to or less than 0.95 with a 95% probability at a 95% confidence level. Reactivity effects of abnormal and accident conditions have also been evaluated to assure that under credible abnormal and accident conditions, the reactivity will be maintained less than 0.95. The purpose of the present analysis is to confirm the acceptability of the rack design for a 3 of 4 storage pattern with a blocking device in the empty storage cell location for the designated fuel assembly design. A description of the blocking device is provided in Section 4.0.

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

- *Code of Federal Regulations,* Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling".

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.1, Criticality Safety of Fresh and Spent Fuel Storage and Handling, Rev. 3 – March 2007.
- USNRC Letter of April 14, 1978 to all Power Reactor Licensees – OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications (GL-78-011), including modification letter dated January 18, 1979 (GL-79-004).
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December 1981.
- ANSI/ANS-8.17-1974, 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 from L. Kopp to Timothy Collins, August 19, 1998 (NRC ADAMS Accession # ML0727102480).
- *Code of Federal Regulations,* Title 10, Part 50, Section 68, "Criticality Accident Requirements"

USNRC guidelines and the applicable ANSI standards specify that the maximum *effective* multiplication factor, k_{eff}, including uncertainties, shall be less than or equal to 0.95. The *infinite* multiplication factor, k_{inf}, is calculated here for a radially and axially infinite array, neglecting neutron loss due to leakage from the actual storage rack, and therefore is a higher and more conservative value.

4.0 ASSUMPTIONS

To assure that the true reactivity will always be less than the calculated reactivity, the following conservative assumptions were made:

- The racks were assumed to contain the most reactive fuel authorized to be stored.
- Moderator in the spent fuel pool rack is pure, unborated water at a temperature that bounds both normal and accident temperatures, corresponding to the highest reactivity.
- Criticality safety analyses are based upon the infinite multiplication factor (k_{inf}), i.e., lattice of storage racks is assumed infinite in all directions. No credit is taken for axial or radial neutron leakage, except in the assessment of certain abnormal or accident conditions where neutron leakage is inherent.
- Neutron absorption in minor structural members is neglected, i.e. spacer grids are replaced by water.
- The Boraflex is replaced with water. This assumption neglects any residual poison that may still be in the racks.

- The 3 of 4 storage configuration requires blocking devices to be placed into those storage cells intended to remain empty. The blocking device is assumed to be a schedule 10, 5" diameter aluminum pipe. The parameters of the aluminum pipe are provided in Table 5.2. Alternatively, a BWR channel is acceptable for use as a blocking device.

5.0 INPUT DATA

5.1 Fuel Assembly Specifications

The spent fuel storage racks are designed to accommodate BWR fuel assemblies from both Unit 1 and Unit 2 of the LaSalle nuclear power station. The design specification for the ATRIUM-10 fuel assembly, which was used for this analysis, is given in Table 5.1. Exelon specified an ATRIUM-10 fuel assembly with a reactivity equivalent uniform enrichment of 2.45 wt% ^{235}U. The equivalent fresh fuel enrichment has been determined in the rack geometry, to bound all past and present fuel assemblies of any type in both LaSalle Unit 1 and Unit 2, in terms of reactivity in the racks.

5.2 Storage Rack Cell Specifications

The storage cell characteristics of the BWR racks that were used in the criticality evaluations are summarized in Table 5.2. A blocking device is required for the storage cells that are required to remain empty. Dimensions and materials of the blocking device are provided in Table 5.2.

6.0 COMPUTER CODES

In the fuel-rack evaluation, criticality analysis of the high-density spent fuel storage racks were performed with the MCNP-4A [5] code, a three-dimensional continuous energy Monte Carlo code. Independent verification calculations were made with the CASMO-4 code [1].

Benchmark calculations are presented in Appendix A of this report and indicate a bias of 0.0009 ± 0.0011 for MCNP-4A. In the geometric model used in the calculations, each fuel rod and its cladding were explicitly described and reflecting boundary conditions were used in the axial direction and periodic boundary conditions were used at the equivalent centerline between storage cells. These boundary conditions have the effect of conservatively creating an infinite array of storage cells in all directions.

The MCNP-4A computer code was used as the primary method of analysis, because it is capable of properly addressing the geometric configuration to be analyzed (3 of 4 storage). MCNP-4A was also used to assess the reactivity consequences of eccentric fuel positioning and other conditions that required a three-dimensional model.

7.0 CALCULATIONS

This section will describe the calculations that were used to determine the acceptable storage criteria for the BWR racks in the LaSalle Unit 2 spent fuel pool. Unless otherwise stated, all calculations assumed nominal characteristics for the fuel and the fuel storage cells. The effect of the manufacturing tolerances is accounted for with a reactivity adjustment as described below.

Figure 5.1 is a plot of the calculational model used in MCNP-4A. Figure 5.1 was created with the two dimensional plotter in MCNP-4A and clearly indicates the explicit modeling of the fuel rods in each assembly.

The goal of the BWR calculations was to verify that the fuel assemblies listed in Table 5.1 is acceptable for storage with maximum planar uniform enrichment less than or equal to a reactivity equivalent enrichment of 2.45 wt% ^{235}U. An equivalent reactivity, fresh, no gadolinia fuel assembly was determined that provided a bounding reactivity to the maximum reactivity lattice at its peak reactivity exposure.

7.1 Manufacturing Tolerances

In the calculation of the final k_{inf}, the effect of manufacturing tolerances on reactivity must be included. CASMO-4 was used to perform these calculations. The reference fuel assembly, with an uniform enrichment of 2.45 wt% ^{235}U was used for these studies. To determine the Δk associated with a specific manufacturing tolerance, the reference k_{inf} was compared to the k_{inf} from a calculation with the positive and negative value of the tolerance included. Note that for the individual parameters associated with a tolerance, no statistical approach is utilized. Instead, the full tolerance value is utilized to determine the maximum reactivity effect. All of the positive Δk values from the various tolerances are statistically combined (square root of the sum of the squares) to determine the final reactivity allowance for manufacturing tolerances. Only the Δk values in the positive direction (increasing reactivity) were used in the statistical combination.

The following is a list of the manufacturing tolerances that were included.

- Fuel Rod Pitch[2] - ± 0.005 inches
- Cell box ID - ± 0.02 inches.

Other manufacturing tolerances of the fuel assembly and the rack were provided separately and include the reactivity effect of manufacturing tolerances for the following parameters:

- UO_2 density
- Enrichment
- Box wall thickness
- Storage Cell Pitch
- Channel Bulge

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[2] The fuel rod pitch tolerance is based on the tolerance over the width of the assembly.

- Pellet diameter
- Cladding diameter
- Pellet void volume
- Gadolina content

Table 7.2 shows the Δk from the reference k_{inf} as compared to the k_{inf} from the cases with the manufacturing tolerances included. The Δk for fuel rod pitch and storage cell inner dimension were calculated for a fresh assembly of uniform 2.45 wt% enrichment. The Δk for the other rack and fuel manufacturing tolerances were calculated for the bounding ATRIUM-10 lattice. These calculations are performed for an infinite array of BWR storage cells ensuring that the calculated reactivity effect from the manufacturing tolerances are conservative compared to the 3 of 4 storage configuration.

7.2 Temperature Effect

The effect on reactivity of varying the spent fuel pool temperature was evaluated using CASMO-4. The results are presented in Table 7.3. The highest temperature evaluated was 123°C (254° F). A case including 10% void was also evaluated at this temperature in order to simulate boiling at the bottom of the spent fuel pool. These results clearly indicate that the spent fuel pool temperature coefficient of reactivity is positive, with additional voids reducing the reactivity. Therefore, all design basis calculations are performed at the maximum temperature of 123° C, with no voids. Because the MCNP-4A calculations are valid at 300K (27° C) the difference in reactivity between 27° C and 123° C is applied as a bias in the final calculation of the maximum k_{eff}.

7.3 Effect of the Channel and Eccentric Fuel Positioning

7.3.1 Channel Removal and Channel Thickness

The BWR fuel assemblies usually have a zircaloy channel attached to the fuel bundle. However, it can not be guaranteed that this channel will be present during storage. Therefore, MCNP-4A calculations were performed to verify that including the channel in the final analysis is conservative.

Additionally, the channels do not have a uniform thickness around the entire assembly. Some channels are typically thinner on the sides and thicker on the corners. To reduce the complexity of the model, the MCNP-4A and CASMO-4 models assume that the channels are uniformly thick and the corners are square (rather than rounded). To ensure that the models are conservative, the effect of the channel thickness on reactivity was determined.

The results of these studies show that by modeling the channel with a uniform maximum thickness, the results are conservative.

7.3.2 Eccentric Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell and in the BWR rack there are bottom fittings and spacers that mechanically restrict lateral movement of the fuel assemblies. Nevertheless, MCNP-4A calculations were made with the fuel assemblies assumed to be in the corner of the storage rack cell. These calculations indicate that eccentric positioning results in a decrease in reactivity. The highest reactivity, therefore, corresponds to the reference design with the fuel assemblies positioned at the center of the storage cells.

7.4 Effect of Fuel Assembly Orientation

All calculations were performed with the ATRIUM-10 fuel assemblies oriented identically, as indicated in Figure 5.1. Since the ATRIUM-10 fuel assembly is not symmetric with respect to the center of the assembly, additional calculations were performed to determine if any reactivity effect is associated with the orientation of the fuel assemblies in the rack storage cells. Nine different orientations of the three ATRIUM-10 fuel assemblies were modeled. The results of these calculations show that there is no statistically significant increase in reactivity due to different orientations of the fuel assemblies.

7.5 Maximum k_{eff}

Using the calculational model shown in Figure 5.1 and the reference ATRIUM-10 fuel assembly, the k_{eff} in the LaSalle Unit 2 BWR storage racks has been calculated with MCNP-4A. The determination of the maximum k_{eff}, which is based on the formula in Section 2, is calculated in Table 7.1. Table 7.1 summarizes the results and demonstrates that by limiting the equivalent enrichment for the LaSalle fuel assemblies the k_{inf} in the spent fuel storage racks with a 3 of 4 storage configuration and taking no credit for Boraflex will be less then 0.95.

7.6 Long Term Reactivity Changes

At reactor shutdown, the reactivity of the fuel initially decreases due to the growth of ^{135}Xe, from ^{135}I decay. Subsequently, the Xenon decays and the reactivity increases to a maximum at several hundred hours when the Xenon is gone. Over the next 30 years, the reactivity continuously decreases due primarily to ^{241}Pu decay and ^{241}Am growth. At lower burnup, the reactivity decrease will be less pronounced since less ^{241}Pu would have been produced. No credit is taken for this long-term decrease in reactivity other than to indicate additional and increasing conservatism in the design criticality analysis.

7.7 Abnormal and Accident Conditions

7.7.1 Dropped Fuel Assembly

For a drop on top of the rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel region of more then 12 inches, which is sufficient to preclude neutron coupling (i.e. an effectively infinite separation).

It is also possible to vertically drop an assembly into a location occupied by another assembly or a blocking device. Such a vertical impact on an assembly would at most cause a small compression of the stored assembly, reducing the water-to-fuel ratio and thereby reducing reactivity. In addition, the distance between the active fuel regions of both assemblies will be more than sufficient to ensure no neutron interaction between the two assemblies. A vertical impact on a blocking device would cause a small amount of buckling in the blocking device itself. The distance between the active fuel region of the dropped assembly and the active fuel region of the surrounding stored assemblies must remain sufficiently large to prevent an inadvertent criticality.

The last scenario is the drop of a fuel assembly into an open storage cell. The dropped assembly would impact the baseplate and could result in a localized deformation of the baseplate that would affect that storage cell and the cells immediately surrounding it. The consequence of this drop accident on criticality is that the active fuel length of that fuel assembly, and possibly the surrounding assemblies, could extend below the active length of the remaining assemblies. The misalignment of the active fuel regions of adjacent fuel assemblies would lead to more neutron leakage and a corresponding reduction in reactivity.

7.7.2 Fuel Rack Lateral Movement

With no consideration of the Boraflex panels and all calculations performed for an infinite array of storage cells, the maximum reactivity of the storage rack is not dependent upon the water gap spacing between modules. Thus, misalignment of the racks or seismically induced movement will not affect the reactivity of the rack. However calculations were performed for a 6x6 array of storage cells with the 3 rows shifted by one cell with respect to the other 3 rows. Results of the calculation are provided in Table 7.4 and shows that the reactivity effect of the lateral movement of the storage racks is negative.

7.7.3 Abnormal Location of a Fuel Assembly

It is hypothetically possible to suspend a fuel assembly of the highest allowable reactivity outside and adjacent to the fuel rack, although such an accident condition is highly unlikely. The exterior walls of the rack modules facing the outside (where such an accident condition might be conceivable) is a region of high neutron leakage. The worst case would occur if an assembly were mislocated outside of the rack and facing rack storage cells filled with design basis fuel on two face adjacent sides, i.e in a corner between the rack and the spent fuel pool wall. Calculations were performed for the above described geometry to determine the reactivity effect of a misplaced assembly outside the rack. Results of the calculation are provided in Table 7.4 and show that the misplacement of a fuel assembly outside the racks does not cause an increase in reactivity.

7.8 Misloading of a Fuel Assembly in a Location Intended to be Empty

The 3 of 4 storage configuration requires blocking devices to be placed into those storage cells intended to remain empty. Due to the stringent administrative controls of placing the blocking devices, it is not considered credible that a fuel assembly could be inadvertently loaded into one of the storage cells intended for a blocking device. However calculations were performed for a 6x6 array of storage cells with the central blocking device replaced with a fuel assembly. Results of the calculation are provided in Table 7.5 and shows that the reactivity of the rack with the misloading of a fuel assembly in a storage cell intended to contain a blocking device remains subcritical.

7.9 Interfaces Between Areas With and Without Boraflex Degradation

The analysis described above assumes that all storage cell locations exhibit an unacceptable level of degradation and therefore no credit is taken for the residual Boraflex. However, in reality there are areas in the LaSalle spent fuel pool where the Boraflex has not degraded beyond acceptable levels. These areas will continue to store spent fuel in a full 4-of-4 array in accordance with the existing licensing criteria. For interfaces between areas of 3-of-4 storage and 4-of-4 storage the following operational controls must be adhered to:

- Each cluster of 4 storage cells (2x2) must meet either the criteria for 4-of-4 storage or the criteria of 3-of-4 storage.
- In each cluster of 4 storage cells (2x2), if one storage cell is considered degraded (one or more of the four surrounding Boraflex panels is considered degraded), then one of the four cells must contain a blocking device.

These operational controls will ensure that the spent fuel pool remains within an existing analyzed condition.

8.0 REFERENCES

[1] M. Edenius, et al., "CASMO-4, A Fuel Assembly Burnup Program, User Manual", Studsvik/SOA-95/1, Studsvik of America, Inc., and Studsvik Core Analysis AB (proprietary).

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

[3] D. Knott, "CASMO-4 Benchmark Against Critical Experiments", SAO-94/13, Studsvik of America, Inc., (proprietary).

[4] D. Knott, "CASMO-4 Benchmark against MCNP," SOA-94-12, Studsvik of America, Inc., (proprietary)

[5] J.F. Briesmeister, Ed., "MCNP – A General Purpose Monte Carlo N-Particle Transport Code, Version 4A", Los Alamos National Laboratory, LA-12625-M (1993).

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

[7] L.I. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Plants", USNRC memorandum, Kopp to Collins, August 1998.

[8] "A Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in burnup Credit Criticality Safety Analyses for PWR Spent Fuel Pool Storage," NUREG/CR-6683, ORNL/TM-2000/230, September 2000.

Table 5.1

BWR Fuel Characteristics

Fuel Assembly	ATRIUM-10
Clad O.D. (in.)	0.3957
Clad I.D. (in.)	0.3480
Clad Material	Zr
Pellet Diameter (in.)	0.3413
Pellet Density (gm/cc) [3]	10.550
Fuel Rod Array	10x10
Number of Fuel Rods	91
Fuel Rod Pitch (in.)	0.510
Number of Water Rods	1 Central Box
Water Rod O.D. (in.)	1.378
Water Rod I.D. (in.)	1.321
Channel I.D. (in.)	5.278
Max Channel Thickness (in.)	0.100

[3] The pellet density is conservatively used as the stack density.

Table 5.2

Fuel Rack Specifications – BWR Boraflex Racks

Parameter	Value
Cell ID, Inches	6.00 ± 0.02
Box Wall Thickness, Inches	0.090 ± 0.009
Cell Pitch[4], Inches	6.255 (6.25 min)
Boraflex Pocket Thickness	0.075
Blocking Device OD, Inches	5.563
Blocking Device Thickness, Inches	0.134[5]
Blocking Device Material	Aluminum

[4] Cell Pitch nominal value is calculated from other storage rack parameters.
[5] A pipe thickness of 0.268 inches was conservatively modeled.

Table 7.1

Summary of the Criticality Safety Analyses for LaSalle Unit 2 BWR Racks

Equivalent Uniform Enrichment [wt% ^{235}U]		2.45
Uncertainties		
Bias Uncertainty (95%/95%)	± 0.0011	
Calculational Statistics (95%/95%, 2.0×σ)	± 0.0014	
Fuel Eccentricity	Negative	
Removal of Flow Channel	Negative	
Rack and Fuel Tolerances	± 0.0111	
Statistical Combination of Uncertainties	± 0.0112	
Reference k_{eff} (MCNP-4A)		0.9261
Biases		
Temperature Bias		0.0080
Calculational Bias (see Appendix A)		0.0009
Maximum k_{eff}		**0.9462**
Regulatory Limiting k_{eff}		**0.9500**

Table 7.2

Reactivity Effect of Manufacturing Tolerances for the LaSalle Unit 2 BWR Racks

Tolerance	**Δk**
Fuel Rod Pitch	0.0007
Cell Inner Dimension	0.0008
Other Rack and Fuel Tolerances[6]	0.0110
Statistical Combination	**0.0111**

[6] Includes the reactivity effect of manufacturing tolerances for the UO_2 density, enrichment, box wall thickness, storage cell pitch, channel bulge, pellet diameter, cladding diameter, pellet void volume and gadolinia content.

Table 7.3

Reactivity Effect of Temperature Variation in the LaSalle Unit 2 BWR Racks

Temperature (°F)	**Δk**
39.2 (4 °C)	-0.0022
68 (20 °C)	-0.0006
80.33 (300K)	Reference
254 (123 °C)	+0.0080
254 + 10% Void	+0.0071

Table 7.4

Reactivity Effect of Abnormal/Accident Conditions in the LaSalle Unit 2 BWR Racks

Condition	**Reactivity Effect (Δk)**
Dropped Fuel Assembly	Negligible
Fuel Rack Movement	-0.0017
Misplaced Assembly Outside Rack	-0.0011

Table 7.5

Reactivity Effect of Non-Credible Accident Conditions in the LaSalle Unit 2 BWR Racks

Condition	**Maximum k_{eff}**
Misloaded Assembly	0.9880

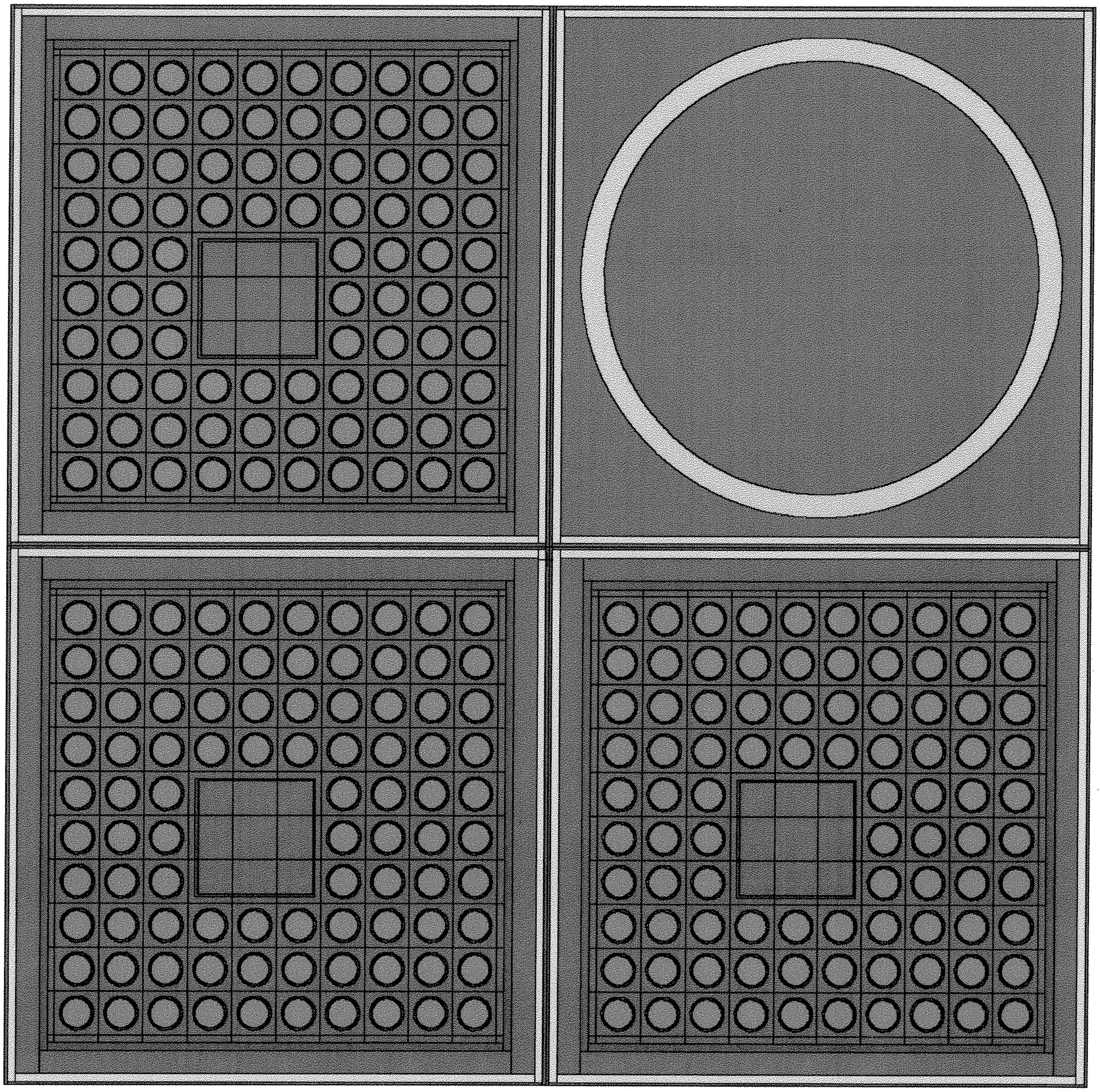

Figure 5.1: A Two Dimensional Representation of the Actual Calculational Model Used For the BWR Rack Analysis. This Figure was Drawn (To Scale) with the Two-Dimensional Plotter in MCNP-4A.

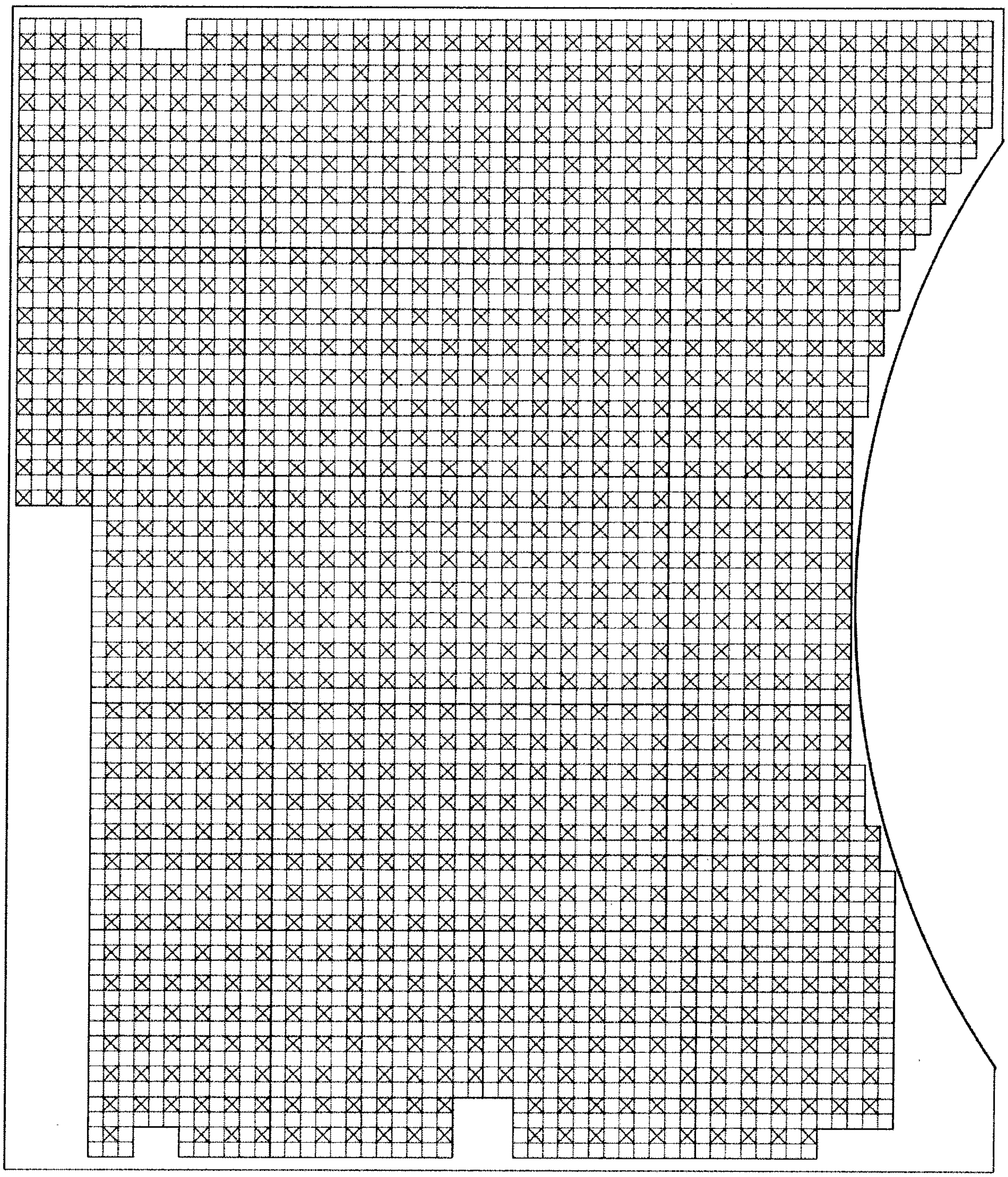

Figure 7.1: LaSalle Unit 2 Spent Fuel Pool Layout for 3 of 4 Configuration[7]

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[7] X's denote the suggested locations for storage cell blocking devices.

Appendix A
Benchmark Calculations

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(this appendix was taken from a different report and because of this the next page is labeled Appendix 4A, Page 1)

APPENDIX 4A: BENCHMARK CALCULATIONS

4A.1 INTRODUCTION AND SUMMARY

Benchmark calculations have been made on selected critical experiments, chosen, in so far as possible, to bound the range of variables in the rack designs. Two independent methods of analysis were used, differing in cross section libraries and in the treatment of the cross sections. MCNP4a [4A.1] is a continuous energy Monte Carlo code and KENO5a [4A.2] uses group-dependent cross sections. For the KENO5a analyses reported here, the 238-group library was chosen, processed through the NITAWL-II [4A.2] program to create a working library and to account for resonance self-shielding in uranium-238 (Nordheim integral treatment). The 238 group library was chosen to avoid or minimize the errors† (trends) that have been reported (e.g., [4A.3 through 4A.5]) for calculations with collapsed cross section sets.

In rack designs, the three most significant parameters affecting criticality are (1) the fuel enrichment, (2) the ^{10}B loading in the neutron absorber, and (3) the lattice spacing (or water-gap thickness if a flux-trap design is used). Other parameters, within the normal range of rack and fuel designs, have a smaller effect, but are also included in the analyses.

Table 4A.1 summarizes results of the benchmark calculations for all cases selected and analyzed, as referenced in the table. The effect of the major variables are discussed in subsequent sections below. It is important to note that there is obviously considerable overlap in parameters since it is not possible to vary a single parameter and maintain criticality; some other parameter or parameters must be concurrently varied to maintain criticality.

One possible way of representing the data is through a spectrum index that incorporates all of the variations in parameters. KENO5a computes and prints the "energy of the average lethargy causing fission" (EALF). In MCNP4a, by utilizing the tally option with the identical 238-group energy structure as in KENO5a, the number of fissions in each group may be collected and the EALF determined (post-processing).

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† Small but observable trends (errors) have been reported for calculations with the 27-group and 44-group collapsed libraries. These errors are probably due to the use of a single collapsing spectrum when the spectrum should be different for the various cases analyzed, as evidenced by the spectrum indices.

Figures 4A.1 and 4A.2 show the calculated k_{eff} for the benchmark critical experiments as a function of the EALF for MCNP4a and KENO5a, respectively (UO_2 fuel only). The scatter in the data (even for comparatively minor variation in critical parameters) represents experimental error[†] in performing the critical experiments within each laboratory, as well as between the various testing laboratories. The B&W critical experiments show a larger experimental error than the PNL criticals. This would be expected since the B&W criticals encompass a greater range of critical parameters than the PNL criticals.

Linear regression analysis of the data in Figures 4A.1 and 4A.2 show that there are no trends, as evidenced by very low values of the correlation coefficient (0.13 for MCNP4a and 0.21 for KENO5a). The total bias (systematic error, or mean of the deviation from a k_{eff} of exactly 1.000) for the two methods of analysis are shown in the table below.

Calculational Bias of MCNP4a and KENO5a	
MCNP4a	0.0009 ± 0.0011
KENO5a	0.0030 ± 0.0012

The bias and standard error of the bias were derived directly from the calculated k_{eff} values in Table 4A.1 using the following equations[††], with the standard error multiplied by the one-sided K-factor for 95% probability at the 95% confidence level from NBS Handbook 91 [4A.18] (for the number of cases analyzed, the K-factor is ~2.05 or slightly more than 2).

$$\bar{k} = \frac{1}{n} \sum_{i}^{n} k_i \tag{4A.1}$$

† A classical example of experimental error is the corrected enrichment in the PNL experiments, first as an addendum to the initial report and, secondly, by revised values in subsequent reports for the same fuel rods.

†† These equations may be found in any standard text on statistics, for example, reference [4A.6] (or the MCNP4a manual) and is the same methodology used in MCNP4a and in KENO5a.

$$\sigma_{\bar{k}}^2 = \frac{\sum_{i=1}^{n} k_i^2 - (\sum_{i=1}^{n} k_i)^2 / n}{n\ (n-1)} \tag{4A.2}$$

$$Bias = (1 - \bar{k}\ ) \pm K\ \sigma_{\bar{k}} \tag{4A.3}$$

where k_i are the calculated reactivities of n critical experiments; $\sigma_{\bar{k}}$ is the unbiased estimator of the standard deviation of the mean (also called the standard error of the bias (mean)); K is the one-sided multiplier for 95% probability at the 95% confidence level (NBS Handbook 91 [4A.18]).

Formula 4.A.3 is based on the methodology of the National Bureau of Standards (now NIST) and is used to calculate the values presented on page 4.A-2. The first portion of the equation, (1- $\bar{k}$), is the actual bias which is added to the MCNP4a and KENO5a results. The second term, $K\sigma_{\bar{k}}$, is the uncertainty or standard error associated with the bias. The K values used were obtained from the National Bureau of Standards Handbook 91 and are for one-sided statistical tolerance limits for 95% probability at the 95% confidence level. The actual K values for the 56 critical experiments evaluated with MCNP4a and the 53 critical experiments evaluated with KENO5a are 2.04 and 2.05, respectively.

The bias values are used to evaluate the maximum k_{eff} values for the rack designs. KENO5a has a slightly larger systematic error than MCNP4a, but both result in greater precision than published data [4A.3 through 4A.5] would indicate for collapsed cross section sets in KENO5a (SCALE) calculations.

4A.2 Effect of Enrichment

The benchmark critical experiments include those with enrichments ranging from 2.46 w/o to 5.74 w/o and therefore span the enrichment range for rack designs. Figures 4A.3 and 4A.4 show the calculated k_{eff} values (Table 4A.1) as a function of the fuel enrichment reported for the critical experiments. Linear regression analyses for these data confirms that there are no trends, as indicated by low values of the correlation coefficients (0.03 for MCNP4a and 0.38 for KENO5a). Thus, there are no corrections to the bias for the various enrichments.

As further confirmation of the absence of any trends with enrichment, a typical configuration was calculated with both MCNP4a and KENO5a for various enrichments. The cross-comparison of calculations with codes of comparable sophistication is suggested in Reg. Guide 3.41. Results of this comparison, shown in Table 4A.2 and Figure 4A.5, confirm no significant difference in the calculated values of k_{eff} for the two independent codes as evidenced by the 45° slope of the curve. Since it is very unlikely that two independent methods of analysis would be subject to the same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias.

4A.3 Effect of ^{10}B Loading

Several laboratories have performed critical experiments with a variety of thin absorber panels similar to the Boral panels in the rack designs. Of these critical experiments, those performed by B&W are the most representative of the rack designs. PNL has also made some measurements with absorber plates, but, with one exception (a flux-trap experiment), the reactivity worth of the absorbers in the PNL tests is very low and any significant errors that might exist in the treatment of strong thin absorbers could not be revealed.

Table 4A.3 lists the subset of experiments using thin neutron absorbers (from Table 4A.1) and shows the reactivity worth (Δk) of the absorber.†

No trends with reactivity worth of the absorber are evident, although based on the calculations shown in Table 4A.3, some of the B&W critical experiments seem to have unusually large experimental errors. B&W made an effort to report some of their experimental errors. Other laboratories did not evaluate their experimental errors.

To further confirm the absence of a significant trend with ^{10}B concentration in the absorber, a cross-comparison was made with MCNP4a and KENO5a (as suggested in Reg. Guide 3.41). Results are shown in Figure 4A.6 and Table 4A.4 for a typical geometry. These data substantiate the absence of any error (trend) in either of the two codes for the conditions analyzed (data points fall on a 45° line, within an expected 95% probability limit).

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† The reactivity worth of the absorber panels was determined by repeating the calculation with the absorber analytically removed and calculating the incremental (Δk) change in reactivity due to the absorber.

4A.4 Miscellaneous and Minor Parameters

4A.4.1 Reflector Material and Spacings

PNL has performed a number of critical experiments with thick steel and lead reflectors.† Analysis of these critical experiments are listed in Table 4A.5 (subset of data in Table 4A.1). There appears to be a small tendency toward overprediction of k_{eff} at the lower spacing, although there are an insufficient number of data points in each series to allow a quantitative determination of any trends. The tendency toward overprediction at close spacing means that the rack calculations may be slightly more conservative than otherwise.

4A.4.2 Fuel Pellet Diameter and Lattice Pitch

The critical experiments selected for analysis cover a range of fuel pellet diameters from 0.311 to 0.444 inches, and lattice spacings from 0.476 to 1.00 inches. In the rack designs, the fuel pellet diameters range from 0.303 to 0.3805 inches O.D. (0.496 to 0.580 inch lattice spacing) for PWR fuel and from 0.3224 to 0.494 inches O.D. (0.488 to 0.740 inch lattice spacing) for BWR fuel. Thus, the critical experiments analyzed provide a reasonable representation of power reactor fuel. Based on the data in Table 4A.1, there does not appear to be any observable trend with either fuel pellet diameter or lattice pitch, at least over the range of the critical experiments applicable to rack designs.

4A.4.3 Soluble Boron Concentration Effects

Various soluble boron concentrations were used in the B&W series of critical experiments and in one PNL experiment, with boron concentrations ranging up to 2550 ppm. Results of MCNP4a (and one KENO5a) calculations are shown in Table 4A.6. Analyses of the very high boron concentration experiments (> 1300 ppm) show a tendency to slightly overpredict reactivity for the three experiments exceeding 1300 ppm. In turn, this would suggest that the evaluation of the racks with higher soluble boron concentrations could be slightly conservative.

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† Parallel experiments with a depleted uranium reflector were also performed but not included in the present analysis since they are not pertinent to the Holtec rack design.

4A.5 MOX Fuel

The number of critical experiments with PuO_2 bearing fuel (MOX) is more limited than for UO_2 fuel. However, a number of MOX critical experiments have been analyzed and the results are shown in Table 4A.7. Results of these analyses are generally above a k_{eff} of 1.00, indicating that when Pu is present, both MCNP4a and KENO5a overpredict the reactivity. This may indicate that calculation for MOX fuel will be expected to be conservative, especially with MCNP4a. It may be noted that for the larger lattice spacings, the KENO5a calculated reactivities are below 1.00, suggesting that a small trend may exist with KENO5a. It is also possible that the overprediction in k_{eff} for both codes may be due to a small inadequacy in the determination of the Pu-241 decay and Am-241 growth. This possibility is supported by the consistency in calculated k_{eff} over a wide range of the spectral index (energy of the average lethargy causing fission).

4A.6 References

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[4A.2] SCALE 4.3, "A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", NUREG-0200 (ORNL-NUREG-CSD-2/U2/R5, Revision 5, Oak Ridge National Laboratory, September 1995.

[4A.3] M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-G Group ENDF/B-Y Cross-Section Library for Use in Criticality Safety Analyses", NUREG/CR-6102 (ORNL/TM-12460) Oak Ridge National Laboratory, September 1994.

[4A.4] W.C. Jordan et al., "Validation of KENOV.a", CSD/TM-238, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, December 1986.

[4A.5] O.W. Hermann et al., "Validation of the Scale System for PWR Spent Fuel Isotopic Composition Analysis", ORNL-TM-12667, Oak Ridge National Laboratory, undated.

[4A.6] R.J. Larsen and M.L. Marx, An Introduction to Mathematical Statistics and its Applications, Prentice-Hall, 1986.

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[4A.9] L.W. Newman et al., Urania Gadolinia: Nuclear Model Development and Critical Experiment Benchmark, BAW-1810, Babcock and Wilcox Company, April 1984.

[4A.10] J.C. Manaranche et al., "Dissolution and Storage Experimental Program with 4.75 w/o Enriched Uranium-Oxide Rods," Trans. Am. Nucl. Soc. 33: 362-364 (1979).

[4A.11] S.R. Bierman and E.D. Clayton, Criticality Experiments with Subcritical Clusters of 2.35 w/o and 4.31 w/o ^{235}U Enriched UO_2 Rods in Water with Steel Reflecting Walls, PNL-3602, Battelle Pacific Northwest Laboratory, April 1981.

[4A.12] S.R. Bierman et al., Criticality Experiments with Subcritical Clusters of 2.35 w/o and 4.31 w/o ^{235}U Enriched UO_2 Rods in Water with Uranium or Lead Reflecting Walls, PNL-3926, Battelle Pacific Northwest Laboratory, December, 1981.

[4A.13] S.R. Bierman et al., Critical Separation Between Subcritical Clusters of 4.31 w/o ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons, PNL-2615, Battelle Pacific Northwest Laboratory, October 1977.

[4A.14] S.R. Bierman, Criticality Experiments with Neutron Flux Traps Containing Voids, PNL-7167, Battelle Pacific Northwest Laboratory, April 1990.

[4A.15] B.M. Durst et al., Critical Experiments with 4.31 wt % ^{235}U Enriched UO_2 Rods in Highly Borated Water Lattices, PNL-4267, Battelle Pacific Northwest Laboratory, August 1982.

[4A.16] S.R. Bierman, Criticality Experiments with Fast Test Reactor Fuel Pins in Organic Moderator, PNL-5803, Battelle Pacific Northwest Laboratory, December 1981.

[4A.17] E.G. Taylor et al., Saxton Plutonium Program Critical Experiments for the Saxton Partial Plutonium Core, WCAP-3385-54, Westinghouse Electric Corp., Atomic Power Division, December 1965.

[4A.18] M.G. Natrella, Experimental Statistics, National Bureau of Standards, Handbook 91, August 1963.

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF † (eV) MCNP4a	EALF † (eV) KENO5a
1	B&W-1484 (4A.7)	Core I	2.46	0.9964 ± 0.0010	0.9898± 0.0006	0.1759	0.1753
2	B&W-1484 (4A.7)	Core II	2.46	1.0008 ± 0.0011	1.0015 ± 0.0005	0.2553	0.2446
3	B&W-1484 (4A.7)	Core III	2.46	1.0010 ± 0.0012	1.0005 ± 0.0005	0.1999	0.1939
4	B&W-1484 (4A.7)	Core IX	2.46	0.9956 ± 0.0012	0.9901 ± 0.0006	0.1422	0.1426
5	B&W-1484 (4A.7)	Core X	2.46	0.9980 ± 0.0014	0.9922 ± 0.0006	0.1513	0.1499
6	B&W-1484 (4A.7)	Core XI	2.46	0.9978 ± 0.0012	1.0005 ± 0.0005	0.2031	0.1947
7	B&W-1484 (4A.7)	Core XII	2.46	0.9988 ± 0.0011	0.9978 ± 0.0006	0.1718	0.1662
8	B&W-1484 (4A.7)	Core XIII	2.46	1.0020 ± 0.0010	0.9952 ± 0.0006	0.1988	0.1965
9	B&W-1484 (4A.7)	Core XIV	2.46	0.9953 ± 0.0011	0.9928 ± 0.0006	0.2022	0.1986
10	B&W-1484 (4A.7)	Core XV ††	2.46	0.9910 ± 0.0011	0.9909 ± 0.0006	0.2092	0.2014
11	B&W-1484 (4A.7)	Core XVI ††	2.46	0.9935 ± 0.0010	0.9889 ± 0.0006	0.1757	0.1713
12	B&W-1484 (4A.7)	Core XVII	2.46	0.9962 ± 0.0012	0.9942 ± 0.0005	0.2083	0.2021
13	B&W-1484 (4A.7)	Core XVIII	2.46	1.0036 ± 0.0012	0.9931 ± 0.0006	0.1705	0.1708

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF[†] (eV) MCNP4a	EALF[†] (eV) KENO5a
14	B&W-1484 (4A.7)	Core XIX	2.46	0.9961 ± 0.0012	0.9971 ± 0.0005	0.2103	0.2011
15	B&W-1484 (4A.7)	Core XX	2.46	1.0008 ± 0.0011	0.9932 ± 0.0006	0.1724	0.1701
16	B&W-1484 (4A.7)	Core XXI	2.46	0.9994 ± 0.0010	0.9918 ± 0.0006	0.1544	0.1536
17	B&W-1645 (4A.8)	S-type Fuel, w/886 ppm B	2.46	0.9970 ± 0.0010	0.9924 ± 0.0006	1.4475	1.4680
18	B&W-1645 (4A.8)	S-type Fuel, w/746 ppm B	2.46	0.9990 ± 0.0010	0.9913 ± 0.0006	1.5463	1.5660
19	B&W-1645 (4A.8)	SO-type Fuel, w/1156 ppm B	2.46	0.9972 ± 0.0009	0.9949 ± 0.0005	0.4241	0.4331
20	B&W-1810 (4A.9)	Case 1 1337 ppm B	2.46	1.0023 ± 0.0010	NC	0.1531	NC
21	B&W-1810 (4A.9)	Case 12 1899 ppm B	2.46/4.02	1.0060 ± 0.0009	NC	0.4493	NC
22	French (4A.10)	Water Moderator 0 gap	4.75	0.9966 ± 0.0013	NC	0.2172	NC
23	French (4A.10)	Water Moderator 2.5 cm gap	4.75	0.9952 ± 0.0012	NC	0.1778	NC
24	French (4A.10)	Water Moderator 5 cm gap	4.75	0.9943 ± 0.0010	NC	0.1677	NC
25	French (4A.10)	Water Moderator 10 cm gap	4.75	0.9979 ± 0.0010	NC	0.1736	NC
26	PNL-3602 (4A.11)	Steel Reflector, 0 separation	2.35	NC	1.0004 ± 0.0006	NC	0.1018

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF† (eV) MCNP4a	EALF† (eV) KENO5a
27	PNL-3602 (4A.11)	Steel Reflector, 1.321 cm sepn.	2.35	0.9980 ± 0.0009	0.9992 ± 0.0006	0.1000	0.0909
28	PNL-3602 (4A.11)	Steel Reflector, 2.616 cm sepn	2.35	0.9968 ± 0.0009	0.9964 ± 0.0006	0.0981	0.0975
29	PNL-3602 (4A.11)	Steel Reflector, 3.912 cm sepn.	2.35	0.9974 ± 0.0010	0.9980 ± 0.0006	0.0976	0.0970
30	PNL-3602 (4A.11)	Steel Reflector, infinite sepn.	2.35	0.9962 ± 0.0008	0.9939 ± 0.0006	0.0973	0.0968
31	PNL-3602 (4A.11)	Steel Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3282
32	PNL-3602 (4A.11)	Steel Reflector, 1.321 cm sepn.	4.306	0.9997 ± 0.0010	1.0012 ± 0.0007	0.3016	0.3039
33	PNL-3602 (4A.11)	Steel Reflector, 2.616 cm sepn.	4.306	0.9994 ± 0.0012	0.9974 ± 0.0007	0.2911	0.2927
34	PNL-3602 (4A.11)	Steel Reflector, 5.405 cm sepn.	4.306	0.9969 ± 0.0011	0.9951 ± 0.0007	0.2828	0.2860
35	PNL-3602 (4A.11)	Steel Reflector, Infinite sepn. ††	4.306	0.9910 ± 0.0020	0.9947 ± 0.0007	0.2851	0.2864
36	PNL-3602 (4A.11)	Steel Reflector, with Boral Sheets	4.306	0.9941 ± 0.0011	0.9970 ± 0.0007	0.3135	0.3150
37	PNL-3926 (4A.12)	Lead Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3159
38	PNL-3926 (4A.12)	Lead Reflector, 0.55 cm sepn.	4.306	1.0025 ± 0.0011	0.9997 ± 0.0007	0.3030	0.3044
39	PNL-3926 (4A.12)	Lead Reflector, 1.956 cm sepn.	4.306	1.0000 ± 0.0012	0.9985 ± 0.0007	0.2883	0.2930

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF † (eV) MCNP4a	EALF † (eV) KENO5a
40	PNL-3926 (4A.12)	Lead Reflector, 5.405 cm sepn.	4.306	0.9971 ± 0.0012	0.9946 ± 0.0007	0.2831	0.2854
41	PNL-2615 (4A.13)	Experiment 004/032 - no absorber	4.306	0.9925 ± 0.0012	0.9950 ± 0.0007	0.1155	0.1159
42	PNL-2615 (4A.13)	Experiment 030 - Zr plates	4.306	NC	0.9971 ± 0.0007	NC	0.1154
43	PNL-2615 (4A.13)	Experiment 013 - Steel plates	4.306	NC	0.9965 ± 0.0007	NC	0.1164
44	PNL-2615 (4A.13)	Experiment 014 - Steel plates	4.306	NC	0.9972 ± 0.0007	NC	0.1164
45	PNL-2615 (4A.13)	Exp. 009 1.05% Boron-Steel plates	4.306	0.9982 ± 0.0010	0.9981 ± 0.0007	0.1172	0.1162
46	PNL-2615 (4A.13)	Exp. 012 1.62% Boron-Steel plates	4.306	0.9996 ± 0.0012	0.9982 ± 0.0007	0.1161	0.1173
47	PNL-2615 (4A.13)	Exp. 031 - Boral plates	4.306	0.9994 ± 0.0012	0.9969 ± 0.0007	0.1165	0.1171
48	PNL-7167 (4A.14)	Experiment 214R - with flux trap	4.306	0.9991 ± 0.0011	0.9956 ± 0.0007	0.3722	0.3812
49	PNL-7167 (4A.14)	Experiment 214V3 - with flux trap	4.306	0.9969 ± 0.0011	0.9963 ± 0.0007	0.3742	0.3826
50	PNL-4267 (4A.15)	Case 173 - 0 ppm B	4.306	0.9974 ± 0.0012	NC	0.2893	NC
51	PNL-4267 (4A.15)	Case 177 - 2550 ppm B	4.306	1.0057 ± 0.0010	NC	0.5509	NC
52	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 21	20% Pu	1.0041 ± 0.0011	1.0046 ± 0.0006	0.9171	0.8868

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF† (eV) MCNP4a	EALF† (eV) KENO5a
53	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 43	20% Pu	1.0058 ± 0.0012	1.0036 ± 0.0006	0.2968	0.2944
54	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 13	20% Pu	1.0083 ± 0.0011	0.9989 ± 0.0006	0.1665	0.1706
55	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 32	20% Pu	1.0079 ± 0.0011	0.9966 ± 0.0006	0.1139	0.1165
56	WCAP-3385 (4A.17)	Saxton Case 52 PuO2 0.52" pitch	6.6% Pu	0.9996 ± 0.0011	1.0005 ± 0.0006	0.8665	0.8417
57	WCAP-3385 (4A.17)	Saxton Case 52 U 0.52" pitch	5.74	1.0000 ± 0.0010	0.9956 ± 0.0007	0.4476	0.4580
58	WCAP-3385 (4A.17)	Saxton Case 56 PuO2 0.56" pitch	6.6% Pu	1.0036 ± 0.0011	1.0047 ± 0.0006	0.5289	0.5197
59	WCAP-3385 (4A.17)	Saxton Case 56 borated PuO2	6.6% Pu	1.0008 ± 0.0010	NC	0.6389	NC
60	WCAP-3385 (4A.17)	Saxton Case 56 U 0.56" pitch	5.74	0.9994 ± 0.0011	0.9967 ± 0.0007	0.2923	0.2954
61	WCAP-3385 (4A.17)	Saxton Case 79 PuO2 0.79" pitch	6.6% Pu	1.0063 ± 0.0011	1.0133 ± 0.0006	0.1520	0.1555
62	WCAP-3385 (4A.17)	Saxton Case 79 U 0.79" pitch	5.74	1.0039 ± 0.0011	1.0008 ± 0.0006	0.1036	0.1047

Notes: NC stands for not calculated.

† EALF is the energy of the average lethargy causing fission.

†† These experimental results appear to be statistical outliers (>3σ) suggesting the possibility of unusually large experimental error. Although they could justifiably be excluded, for conservatism, they were retained in determining the calculational basis.

Table 4A.2

COMPARISON OF MCNP4a AND KENO5a CALCULATED REACTIVITIES[†]
FOR VARIOUS ENRICHMENTS

Enrichment	Calculated $k_{eff} \pm 1\sigma$	
	MCNP4a	**KENO5a**
3.0	0.8465 ± 0.0011	0.8478 ± 0.0004
3.5	0.8820 ± 0.0011	0.8841 ± 0.0004
3.75	0.9019 ± 0.0011	0.8987 ± 0.0004
4.0	0.9132 ± 0.0010	0.9140 ± 0.0004
4.2	0.9276 ± 0.0011	0.9237 ± 0.0004
4.5	0.9400 ± 0.0011	0.9388 ± 0.0004

† Based on the GE 8x8R fuel assembly.

Table 4A.3

MCNP4a CALCULATED REACTIVITIES FOR
CRITICAL EXPERIMENTS WITH NEUTRON ABSORBERS

Ref.	Experiment		Δk Worth of Absorber	MCNP4a Calculated k_{eff}	EALF † (eV)
4A.13	PNL-2615	Boral Sheet	0.0139	0.9994±0.0012	0.1165
4A.7	B&W-1484	Core XX	0.0165	1.0008±0.0011	0.1724
4A.13	PNL-2615	1.62% Boron-steel	0.0165	0.9996±0.0012	0.1161
4A.7	B&W-1484	Core XIX	0.0202	0.9961±0.0012	0.2103
4A.7	B&W-1484	Core XXI	0.0243	0.9994±0.0010	0.1544
4A.7	B&W-1484	Core XVII	0.0519	0.9962±0.0012	0.2083
4A.11	PNL-3602	Boral Sheet	0.0708	0.9941±0.0011	0.3135
4A.7	B&W-1484	Core XV	0.0786	0.9910±0.0011	0.2092
4A.7	B&W-1484	Core XVI	0.0845	0.9935±0.0010	0.1757
4A.7	B&W-1484	Core XIV	0.1575	0.9953±0.0011	0.2022
4A.7	B&W-1484	Core XIII	0.1738	1.0020±0.0011	0.1988
4A.14	PNL-7167	Expt 214R flux trap	0.1931	0.9991±0.0011	0.3722

†EALF is the energy of the average lethargy causing fission.

Table 4A.4

COMPARISON OF MCNP4a AND KENO5a
CALCULATED REACTIVITIES† FOR VARIOUS ^{10}B LOADINGS

^{10}B, g/cm^2	**Calculated k_{eff} ± 1σ**	
	MCNP4a	**KENO5a**
0.005	1.0381 ± 0.0012	1.0340 ± 0.0004
0.010	0.9960 ± 0.0010	0.9941 ± 0.0004
0.015	0.9727 ± 0.0009	0.9713 ± 0.0004
0.020	0.9541 ± 0.0012	0.9560 ± 0.0004
0.025	0.9433 ± 0.0011	0.9428 ± 0.0004
0.03	0.9325 ± 0.0011	0.9338 ± 0.0004
0.035	0.9234 ± 0.0011	0.9251 ± 0.0004
0.04	0.9173 ± 0.0011	0.9179 ± 0.0004

† Based on a 4.5% enriched GE 8x8R fuel assembly.

Table 4A.5

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH THICK LEAD AND STEEL REFLECTORS[†]

Ref.	Case	E, wt%	Separation, cm	MCNP4a k_{eff}	KENO5a k_{eff}
4A.11	Steel Reflector	2.35	1.321	0.9980±0.0009	0.9992±0.0006
		2.35	2.616	0.9968±0.0009	0.9964±0.0006
		2.35	3.912	0.9974±0.0010	0.9980±0.0006
		2.35	∞	0.9962±0.0008	0.9939±0.0006
4A.11	Steel Reflector	4.306	1.321	0.9997±0.0010	1.0012±0.0007
		4.306	2.616	0.9994±0.0012	0.9974±0.0007
		4.306	3.405	0.9969±0.0011	0.9951±0.0007
		4.306	∞	0.9910±0.0020	0.9947±0.0007
4A.12	Lead Reflector	4.306	0.55	1.0025±0.0011	0.9997±0.0007
		4.306	1.956	1.0000±0.0012	0.9985±0.0007
		4.306	5.405	0.9971±0.0012	0.9946±0.0007

† Arranged in order of increasing reflector-fuel spacing.

Table 4A.6

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH VARIOUS SOLUBLE BORON CONCENTRATIONS

Reference	Experiment	Boron Concentration, ppm	Calculated k_{eff}	
			MCNP4a	KENO5a
4A.15	PNL-4267	0	0.9974 ± 0.0012	-
4A.8	B&W-1645	886	0.9970 ± 0.0010	0.9924 ± 0.0006
4A.9	B&W-1810	1337	1.0023 ± 0.0010	-
4A.9	B&W-1810	1899	1.0060 ± 0.0009	-
4A.15	PNL-4267	2550	1.0057 ± 0.0010	-

Table 4A.7

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH MOX FUEL

Reference	Case†	MCNP4a		KENO5a	
		k_{eff}	EALF††	k_{eff}	EALF††
PNL-5803 [4A.16]	MOX Fuel - Exp. No. 21	1.0041±0.0011	0.9171	1.0046±0.0006	0.8868
	MOX Fuel - Exp. No. 43	1.0058±0.0012	0.2968	1.0036±0.0006	0.2944
	MOX Fuel - Exp. No. 13	1.0083±0.0011	0.1665	0.9989±0.0006	0.1706
	MOX Fuel - Exp. No. 32	1.0079±0.0011	0.1139	0.9966±0.0006	0.1165
WCAP-3385-54 [4A.17]	Saxton @ 0.52" pitch	0.9996±0.0011	0.8665	1.0005±0.0006	0.8417
	Saxton @ 0.56" pitch	1.0036±0.0011	0.5289	1.0047±0.0006	0.5197
	Saxton @ 0.56" pitch borated	1.0008±0.0010	0.6389	NC	NC
	Saxton @ 0.79" pitch	1.0063±0.0011	0.1520	1.0133±0.0006	0.1555

Note: NC stands for not calculated

† Arranged in order of increasing lattice spacing.

†† EALF is the energy of the average lethargy causing fission.

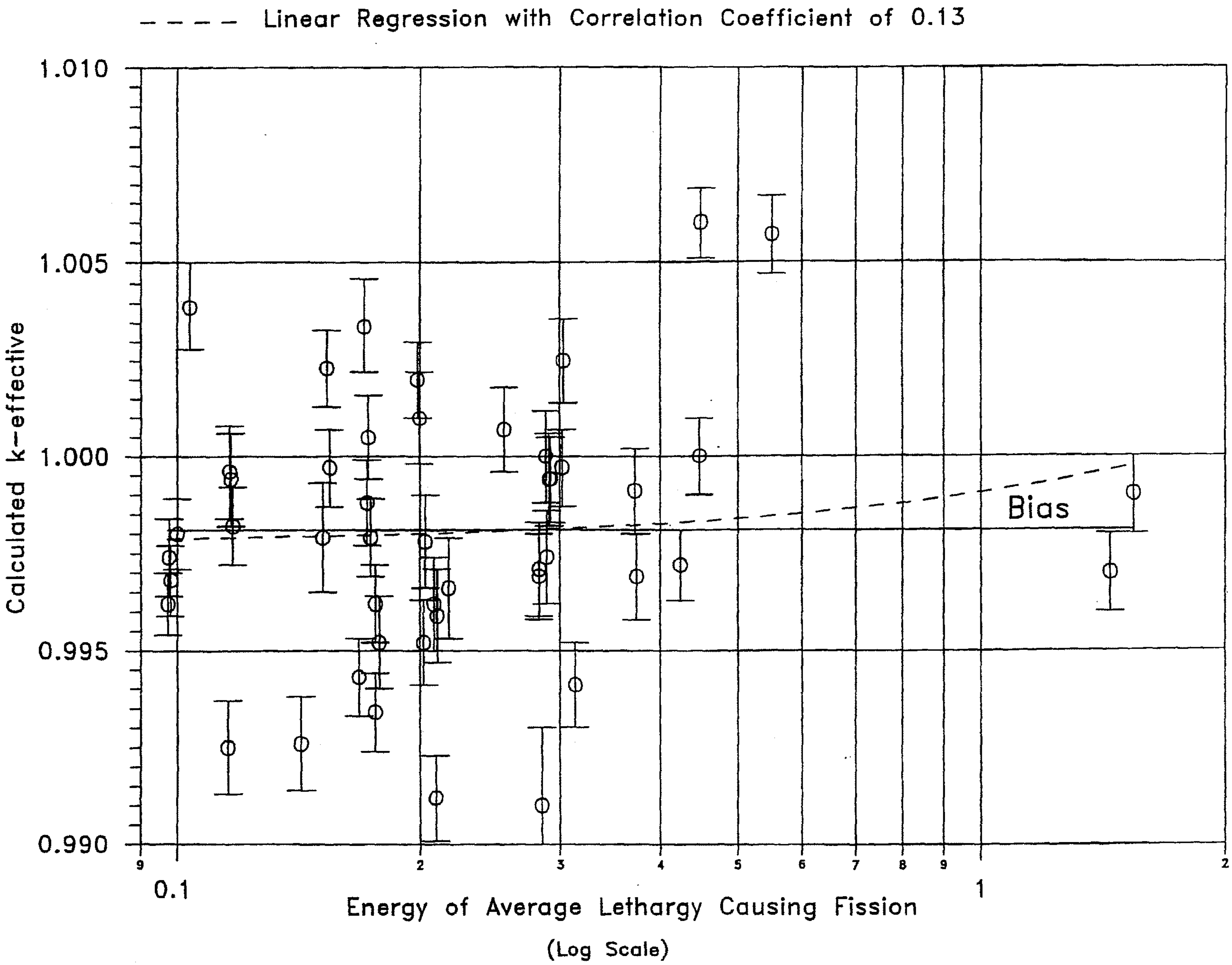

FIGURE 4A.1 MCNP CALCULATED k-eff VALUES for VARIOUS VALUES OF THE SPECTRAL INDEX

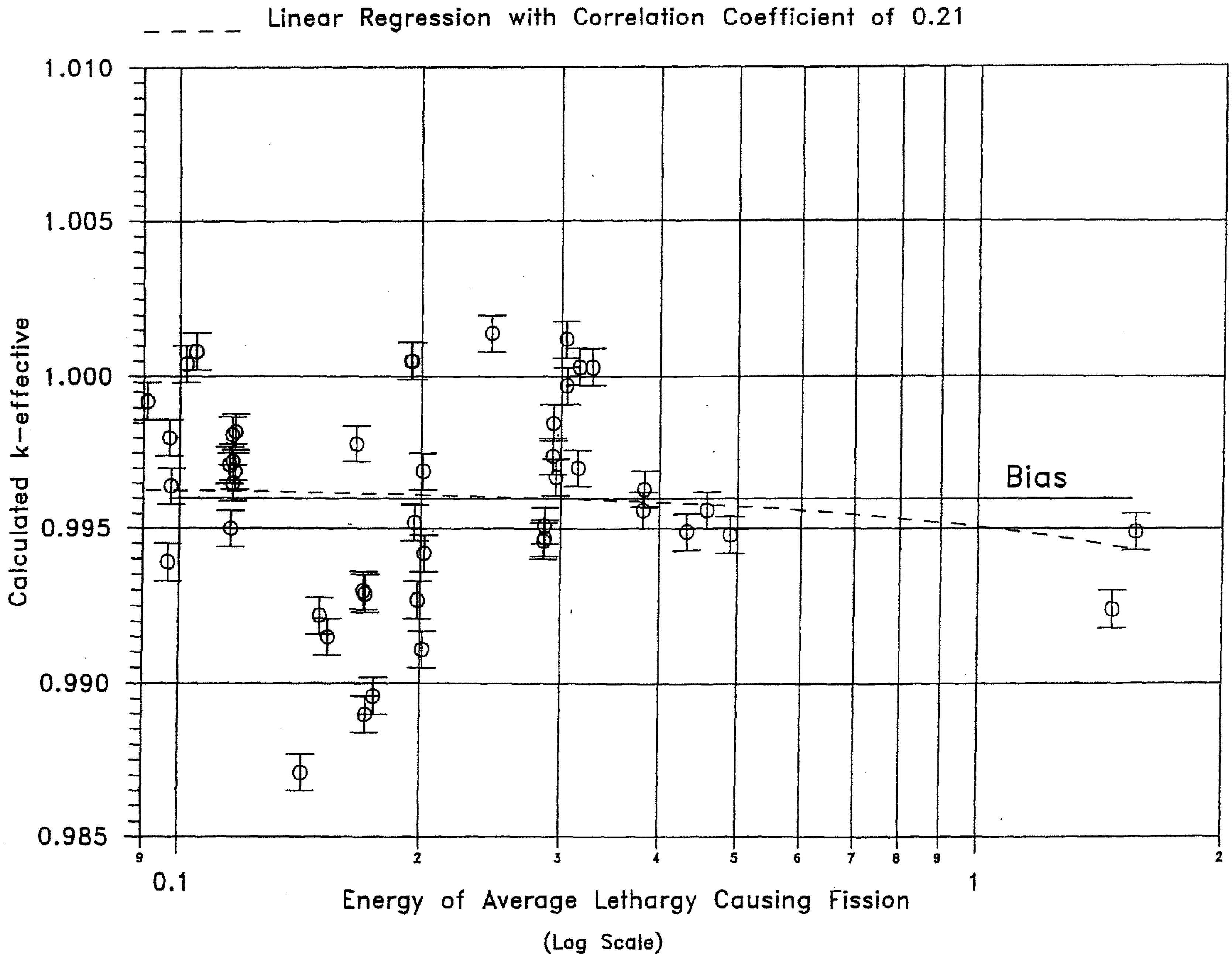

FIGURE 4A.2 KENO5a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

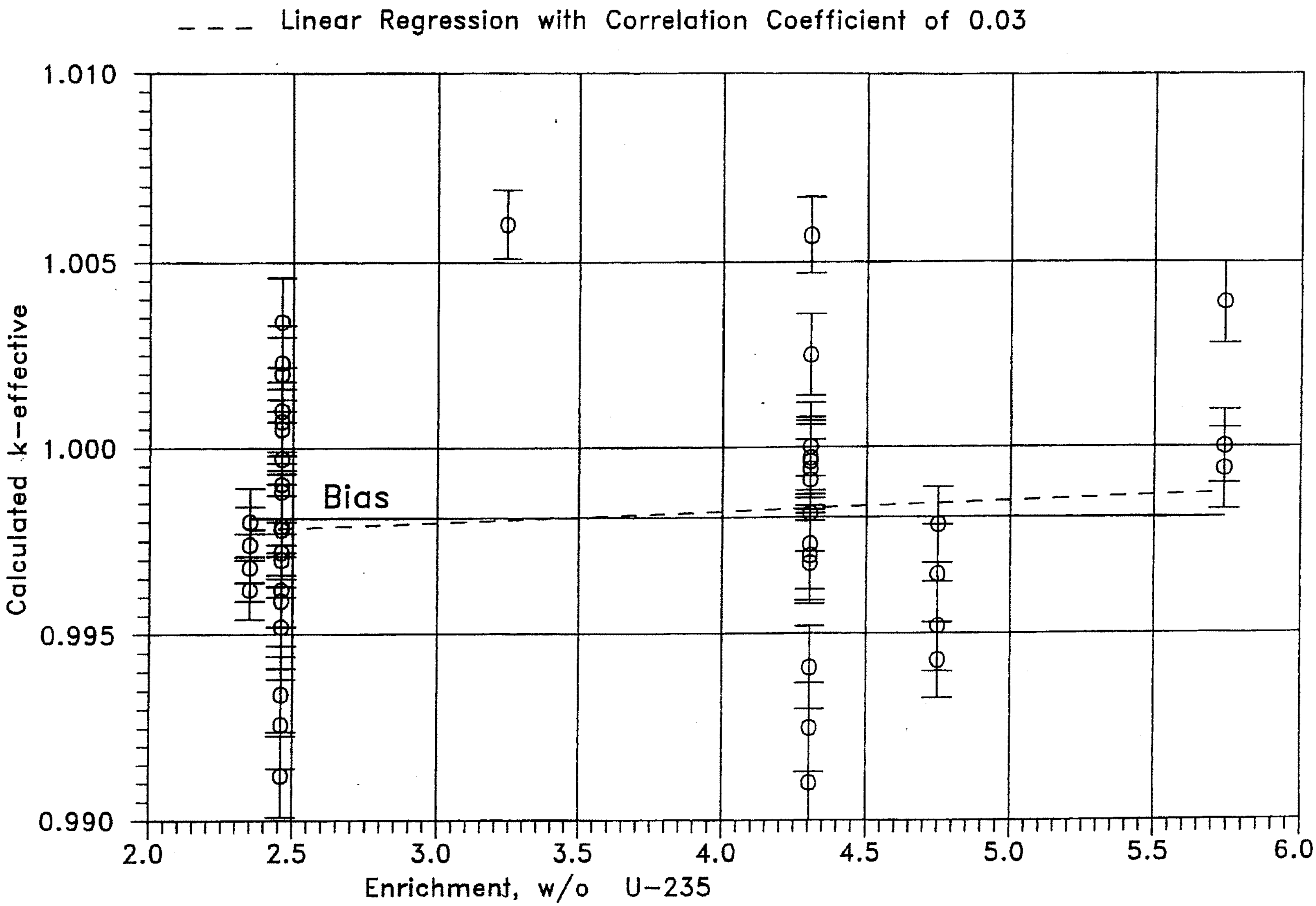

FIGURE 4A.3 MCNP CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

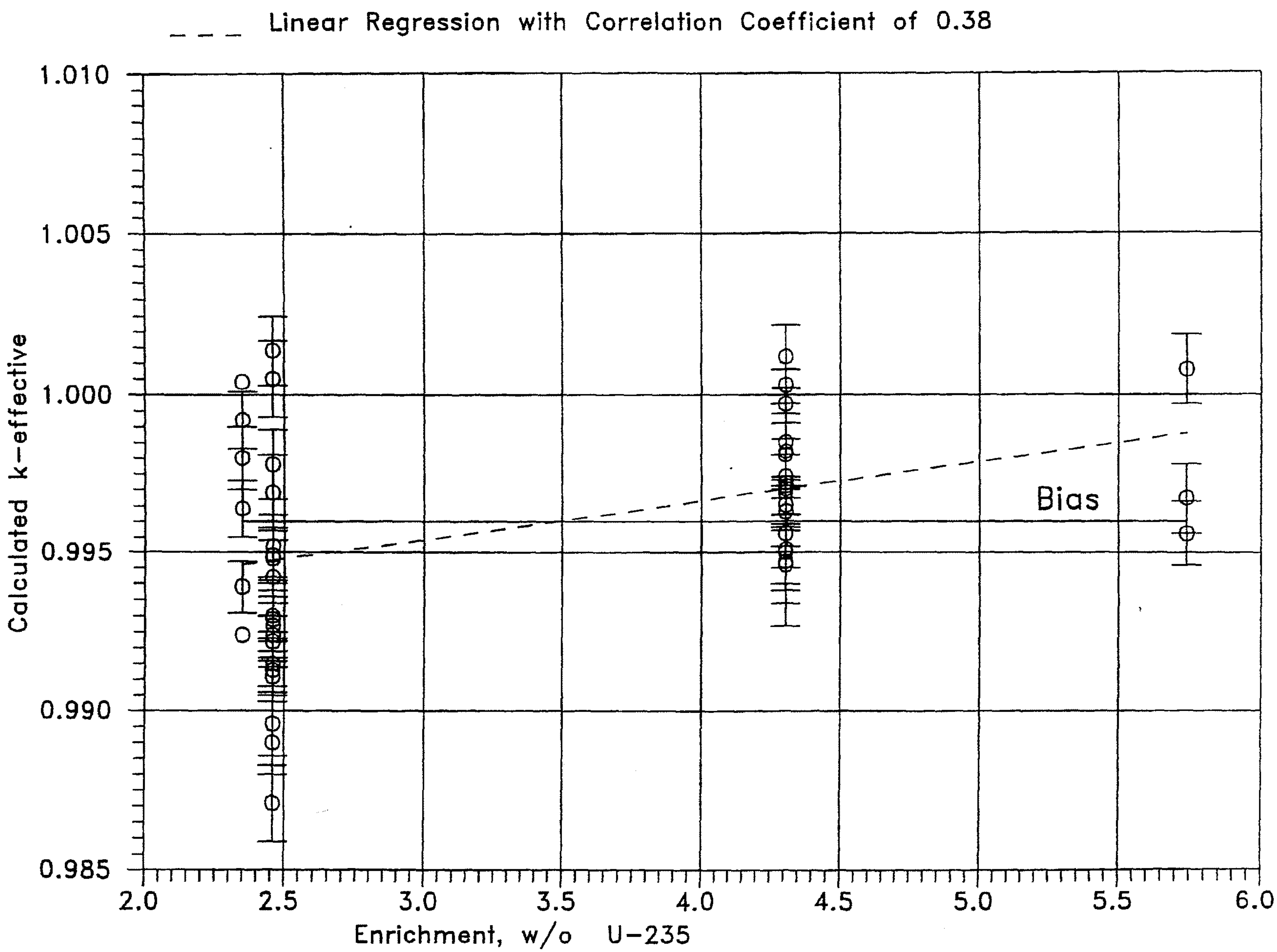

FIGURE 4A.4 KENO CALCULATED k-eff VALUES
AT VARIOUS U-235 ENRICHMENTS

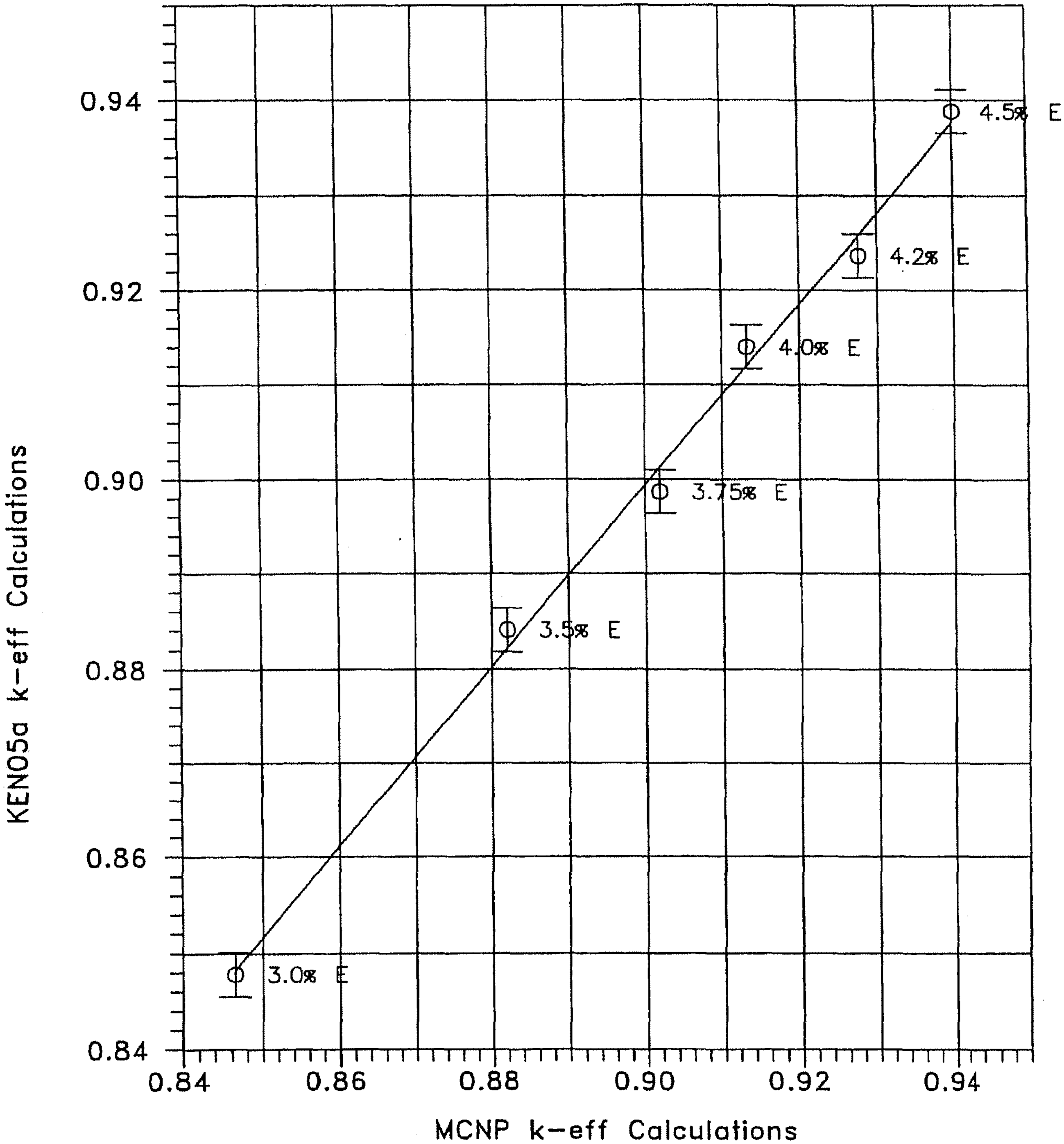

FIGURE 4A.5 COMPARISON OF MCNP AND KENO5A CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

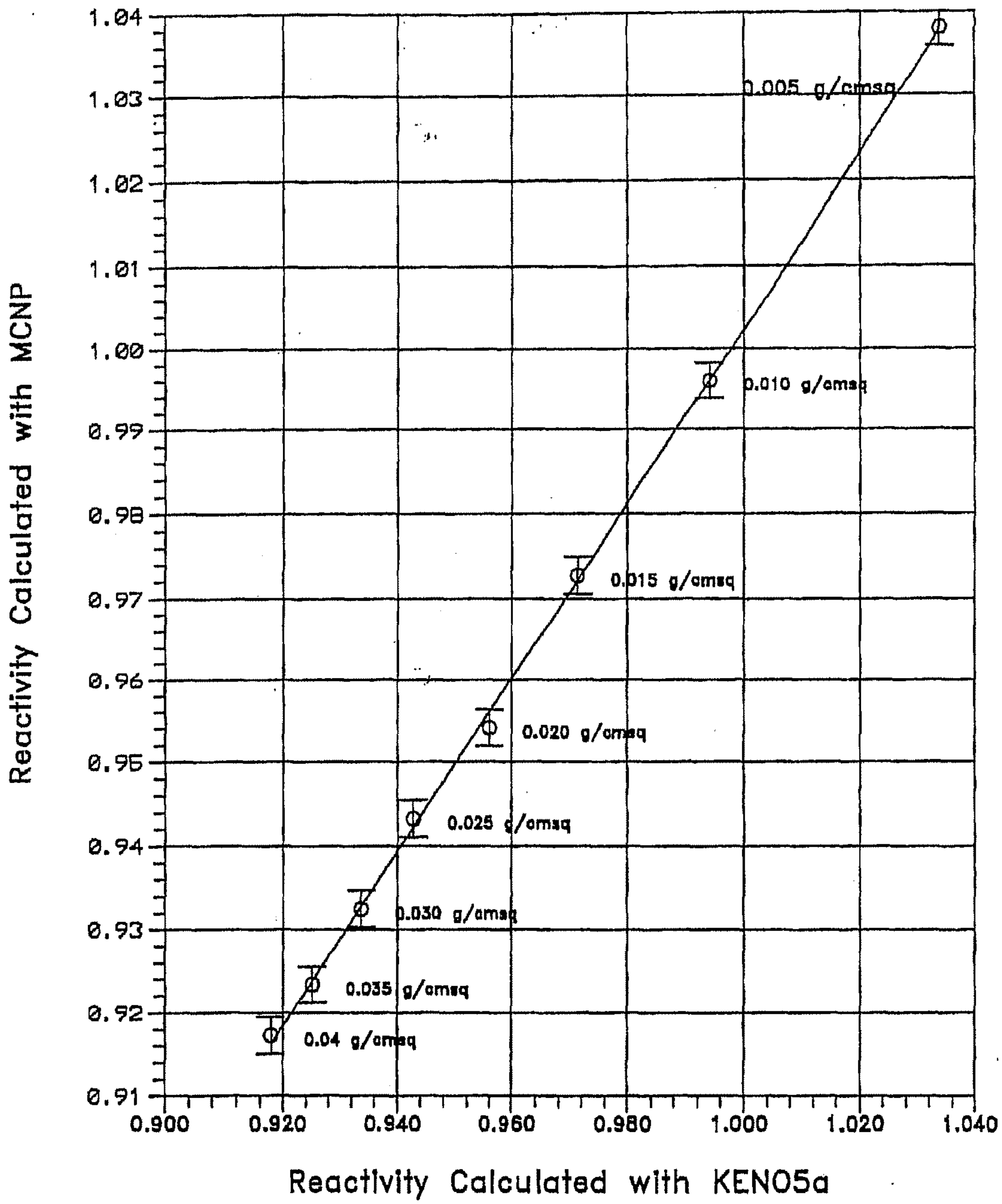

FIGURE 4A.6 COMPARISON OF MCNP AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES

ATTACHMENT 4

AREVA NP Inc. Report No. ANP-2684, "LaSalle Unit 2 Nuclear Power Station Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM^{TM}-10 Fuel in a 2x2-1 Configuration without Boraflex," Revision 0

An AREVA and Siemens company

ANP-2684
Revision 0

LaSalle Unit 2 Nuclear Power Station Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™-10 Fuel in a 2x2-1 Configuration without Boraflex

October 2007

AREVA

AREVA NP Inc.

ANP-2684
Revision 0

LaSalle Unit 2 Nuclear Power Station Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™-10 Fuel in a 2x2-1 Configuration without Boraflex

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AREVA NP Inc.

ANP-2684
Revision 0

Nature of Changes

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

Contents

Tables

Figures

This document contains a total of 26 pages.

Nomenclature

BWR	boiling-water reactor
k-eff	effective neutron multiplication factor
k_∞	infinite lattice neutron multiplication factor
PWR	pressurized water reactor
NRC	Nuclear Regulatory Commission, U.S.
REBOL	reactivity-equivalent at beginning of life

1.0 Introduction

Reference 1 contains an evaluation of the spent fuel storage pool of the LaSalle Unit 2 Nuclear Power Station with AREVA NP Inc.* ATRIUM™-10† fuel assemblies in a repeated 2x2 array with one assembly removed (i.e., 75% checker-board loading) and no credit for Boraflex. The Reference 1 evaluation included the worst credible conditions and uncertainties. This document provides a review of recent ATRIUM-10 fuel assembly design axial lattices relative to those used to define the bounding lattice in Reference 1. This report also summarizes the cases where the reactivity of the Reference 1 bounding lattice has been exceeded. (A list of the LaSalle fuel lattices considered for this evaluation is given in Appendix A).

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

† ATRIUM is a trademark of AREVA NP.

2.0 Summary

The LaSalle Unit 2 spent fuel storage pool criticality safety evaluation performed in Reference 1 can be extended to support ATRIUM-10 fuel assembly designs that meet the criticality safety limits defined in Table 2.1. In addition, the A10B-245L-0G0 REBOL lattice from Reference 1 as modeled by KENO adequately bounds any ATRIUM-10 fuel assembly that meets the criticality safety limits defined in Table 2.1. Finally, a combined statistical uncertainty of 0.00563 Δk_∞ has been calculated to account for LaSalle ATRIUM-10 fuel manufacturing tolerances.

Table 2.1 Criticality Safety Limits for Fuel Assemblies Stored in the LaSalle Unit 2 Spent Fuel Storage Pool

1. ATRIUM-10 Fuel Configuration

Parameter	Nominal ATRIUM-10 Values
Clad OD, in.	.3957
Clad ID, in.	.3480
Pellet Diameter, in.	.3413
Rod Pitch, in.	.510
Fuel Density, % Theoretical for liner fuel	96.26
Water Rods	Internal Channel

2. Fuel may be stored with or without fuel channels

3. Empty locations in the 2x2-1 configuration must preclude a misload condition through physical barriers and/or administrative restrictions.

4. Fuel Design Limitations for Enriched Lattices

Maximum Enrichment, wt% U-235	4.60
Minimum Number of Gd* Rods	11
Minimum wt% Gd_2O_3 in each Gd Rod	6.0

5. ATRIUM-10 fuel assemblies with lattices that do not meet the limitations of item 4 may be stored in the spent fuel pool provided the reactivity of all lattices in the assembly do not exceed a CASMO-4 in-rack k_∞ of 1.0981 at any time during their lifetime (assuming no Boraflex). (The CASMO-4 in-rack geometry to be used for this calculation is shown in Appendix B. The calculation is run at xenon-free conditions with fuel and moderator temperatures at 100 °C).

6. ATRIUM-10 fuel assemblies where all but one lattice meets the item 5 requirement may be stored in the spent fuel pool provided the non-conforming lattice: a) has a zone length of 12" or less, b) is adjacent to the top natural blanket, and c) does not exceed a CASMO-4 in-rack k_∞ of 1.1230 at any time during its lifetime (assuming no Boraflex). (The CASMO-4 in-rack geometry to be used for this calculation is shown in Appendix B. The calculation is run at xenon-free conditions with fuel and moderator temperatures at 100 °C).

7. The spent fuel storage rack design parameters and dimensions are as defined in References 2 and 3.

---

* Gd means gadolinia-bearing (Gd_2O_3) fuel rods.

3.0 Criticality Safety Design Criteria

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

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

B. Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants, issued by the NRC in 1998 (Reference 5).

These documents define the assumptions and acceptance criteria used in this evaluation.

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 ATRIUM-10 mechanical design parameters are summarized in Table 4.1. The assembly design is depicted in Figure 4.1. The LaSalle ATRIUM-10 fuel channel is a uniform wall 0.100-inch-thick channel.

4.2 *Fuel Storage Rack*

The design basis storage rack cell specifications are from References 2 and 3. The calculational model of the storage cell and ATRIUM-10 fuel assembly with a 100-mil channel is shown in Figure 4.2. The original configuration of the storage rack included a neutron absorber material (Boraflex) positioned between the fuel assembly storage cells (see Figure 4.2 of Reference 1); however, this model assumes that the Boraflex has eroded away and has been replaced by water. This rack geometry provides a nominal center-to-center in-rack lattice spacing of 6.255 inches in the non-vertical directions. The 0.09-inch wall thickness stainless steel box, which defines the fuel assembly storage cell, has a nominal inside dimension of 6 inches.

The modeled configuration assumes no Boraflex and one assembly of a repeated 2x2 array is removed (2x2-1). In-rack analyses include ATRIUM-10 lattice configurations with the 0.100-inch-uniform wall fuel channel and with the fuel channel removed. Results demonstrate a negligible difference between the different fuel channel configurations. There are no limitations on the channeling configuration for the ATRIUM-10 assemblies.

Table 4.1 ATRIUM-10 Fuel Assembly Parameters

Fuel Assembly	
Fuel Rod Array	10x10
Fuel Rod Pitch, in.	0.510
Number of Fuel Rods Per Assembly	91
Water Channel	1
Fuel Rods	
Fuel Material	UO_2
Max. Lattice Enrichment, wt% U-235	4.60
Pellet Density, % of Theoretical (liner fuel)	96.26
Pellet Diameter, in.	0.3413
Pellet Void Volume*, %	
Enriched UO_2 (also with Gd_2O_3)	1.2 to 1.4
Natural UO_2	0.9 to 1.2
Cladding Material	Zircaloy-2
Cladding OD, in.	0.3957
Cladding ID, in.	0.3480
Internal Water Channel	
Outside Dimension, in.	1.378
Inside Dimension, in.	1.321
Channel Material	Zircaloy-2 or Zircaloy-4
Fuel Channel (100-mil standard)†	
Outside Dimension	5.478
Inside Dimension	5.278
Channel Material	Zircaloy-2 or Zircaloy-4

* Variations in void volume are not significant in this analysis.

† The conclusions in this report are equally valid for thicker fuel channels.

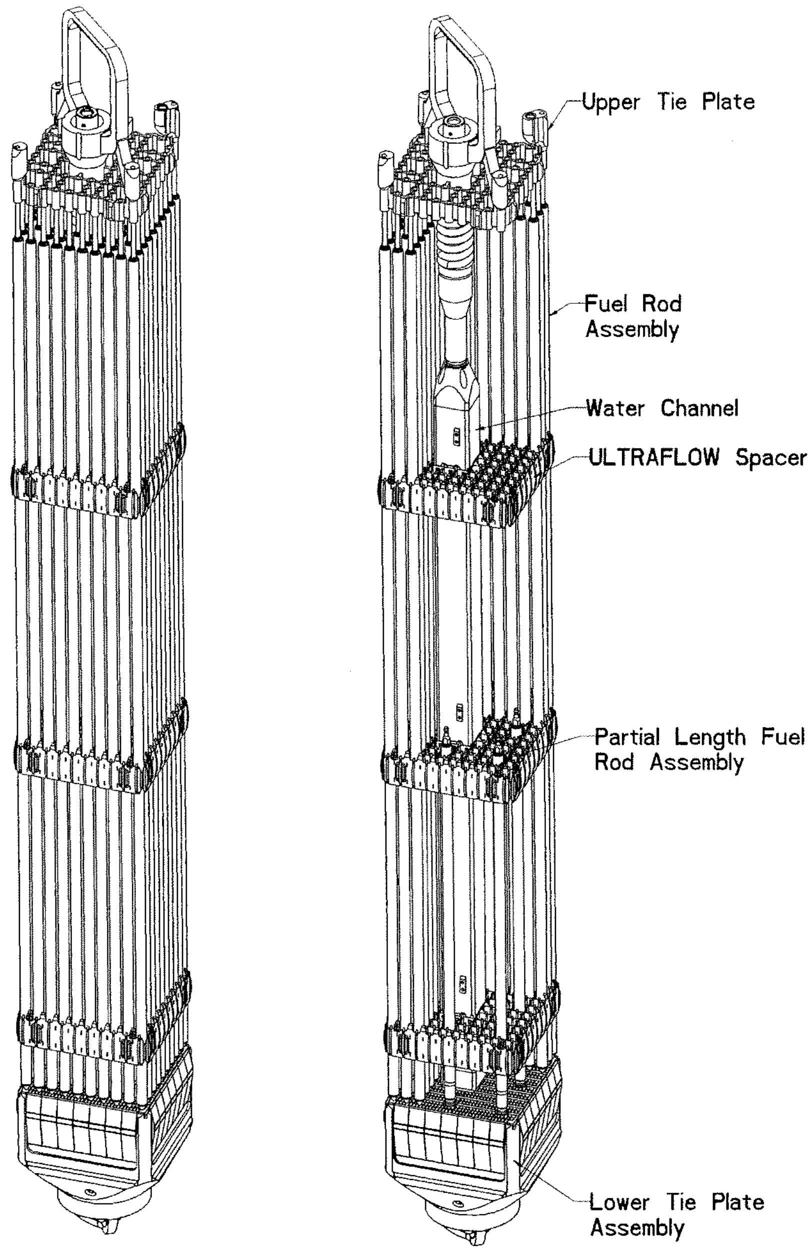

Figure 4.1 Representative ATRIUM-10 Fuel Assembly

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

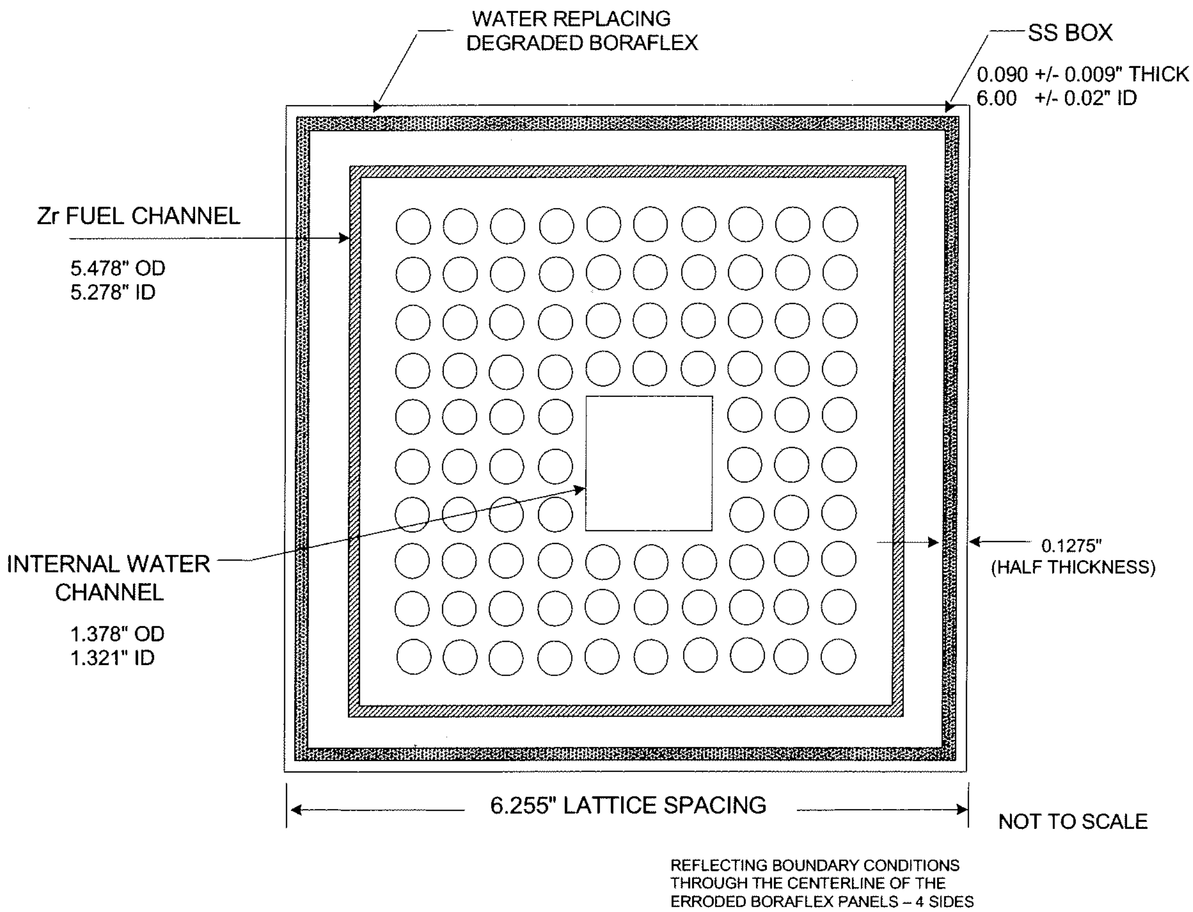

Figure 4.2 Calculational Model of Storage Cell

5.0 Calculation Methodology

The CASMO-4 bundle depletion code (Reference 6) 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 V.a Monte Carlo code, which is part of the SCALE 4.2 Modular Code System (Reference 7). 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 NP has broad experience with both of these codes.

6.0 Criticality Safety Analysis

The Reference 1 criticality safety evaluation uses a single 2.45 wt% U235 REBOL lattice (A10B-245L-0G0) to represent the ATRIUM-10 fuel assembly and demonstrates that the upper limit 95/95 k-eff for the LaSalle Unit 2 spent fuel pool can be met assuming a configuration where one assembly in a repeated 2x2 array is removed and all Boraflex is replaced with water. Recent ATRIUM-10 fuel assembly designs used in the LaSalle reactors have fuel lattices with higher reactivity than the bounding lattice used in Reference 1. A more detailed evaluation has been performed to demonstrate that the single lattice model used in Reference 1 provides sufficient representation of the current ATRIUM-10 fuel assemblies.

6.1 *Fuel Lattice Reactivity Comparison*

An infinite lattice reactivity comparison of all current ATRIUM-10 fuel assemblies has been performed with the CASMO-4 computer code. As summarized in Table 6.1, several top lattices with 4.0 or 4.5 wt% gadolinia have higher k_∞ values than the A10B-460L-11G60 bounding lattice from Reference 1. These high reactivity lattices are all located adjacent to the top natural blanket and have zone lengths of 6" or 12". The A10T-4444L-12G40 lattice has been selected as a secondary bounding lattice and will be represented in KENO using an A10T-270L-0G0 REBOL lattice. All other lattices will be represented in KENO with the appropriate 2.45 wt% U235 REBOL lattice.

6.2 *KENO Geometry Model*

The design basis storage rack is defined in Section 4.2, (Boraflex entirely replaced by water and one assembly in a repeated 2x2 array removed). The ATRIUM-10 fuel assembly model includes 83 full length fuel rods, 8 part length fuel rods, and an internal water channel that occupies the equivalent of 9 fuel rod locations. The full length fuel rods are modeled as: (bottom to top) 6" of natural uranium pellets, 126" of 2.45 wt% U235 pellets, 12" of 2.70 wt% U235 pellets and 5" of natural uranium pellets*. The part length fuel rods are modeled as: (bottom to top) 6" of plenum and 90" of 2.45 wt% U235 pellets. An infinite periodic boundary condition is used in all directions.

* This provides a slightly conservative representation of the 11.00" top natural uranium blankets currently in use with LaSalle fuel assemblies.

6.3 ***KENO Comparison***

The KENO results are listed in Table 6.2 with the explicit two-lattice and blankets geometry model (Case 2) providing a lower k_∞ than the single lattice model (Case 1). It follows from this comparison that the single lattice model used in Reference 1 continues to bound assemblies with more reactive fuel lattices that meet the requirements defined in Table 2.1.

Table 6.1 ATRIUM-10 Fuel Lattice Reactivity Comparison

(Bold font identifies cases that are more reactive than the Reference 1 bounding lattice, Case 1)

Case	Lattice*	Maximum In-Rack k_∞ (CASMO-4)	
		20 °C	100 °C
1	A10B-460L11G60†	1.0894	1.0981
2	A10B-4511L-13G80	1.0547	1.0640
3	A10B-4510L-13G75	1.0614	1.0706
4	A10B-4399L-12G65	1.0735	1.0826
5	A10T-4455L-11G80	1.0612	1.0712
6	A10T-4313L-15G65	1.0566	1.0663
7	**A10T-4409L-10G45**	**1.1125**	**1.1218**
8	**A10T-4400L-10G45**	**1.1131**	**1.1223**
9	A10T-4040L-10G45	1.0863	1.0957
10	**A10T-4444L-12G40**	**1.1136**	**1.1230**
11	**A10T-3986L-12G40**	**1.0939**	**1.1032**
12	REBOL A10B-245L-0G0	1.1001	1.1069
13	REBOL A10T-245L-0G0	1.0916	1.0981
14	REBOL A10T-270L-0G0	1.1219	1.1290

* Note that A10B indicates bottom lattice geometry and A10T indicates top lattice geometry.

† The bounding lattice reactivity values are from Table 6.2 of Reference 1.

Table 6.2 Assembly Model Comparison Results

Fuel Assembly (General)

Physical Description: See Section 4.1

Fuel Channel 100-mil

Storage Cell

Cell Center-to-Center Spacing: 6.255 × 6.255 -inch centers

Boraflex: none (replaced by water)

Special Geometry: 2x2-1 (one empty cell in every 4 locations)

Moderator Temperature: 100°C

Case 1 Fuel Assembly

Physical Description: 2.45 wt% U235

Gadolinia: 0

Geometry: bottom lattice (91 fuel rods)

Case 2 Fuel Assembly

Physical Description: 0.72 wt% U235, (5" at the top and 6" at the bottom)
2.70 wt% U235, (12" adjacent to the top natural blanket)
2.45 wt% U235, (remainder)

Gadolinia: 0

Geometry: actual (91 fuel rods below 96" and 83 fuel rods above 96")

KENO V.a Results

	With fuel channel	
Case	k_∞	σ
1*	0.9165	0.001
2	0.9149	0.001

* From Section 6.6 and Table 6.1 of Reference 1.

7.0 General Uncertainty Conditions

Pellet depletion and fuel manufacturing uncertainties for the ATRIUM-10 fuel are discussed in the following sections. Only the uncertainties with a specific numeric value need to be included in extension calculations.

7.1 *Depletion Uncertainties*

As noted in items 5 and 6 of Table 2.1, all reactivity comparisons are made on a lifetime maximum basis; therefore, no depletion uncertainty is applicable.

7.2 *Burnup Gradient Uncertainties*

Fuel in the reactor core will not receive uniform burn-up across the lattice. This is especially true for fuel assemblies loaded near the edge of a reactor core. For low exposure assemblies the burn-up gradient will generally be small because these assemblies are normally loaded towards the center of the core and higher exposure assemblies will be loaded between them and the core periphery. For high exposure assemblies, the burnup gradient can be larger but it is also of less significance because the assembly has been depleted past its maximum reactivity condition. Because fuel rods near the fuel lattice edge deplete faster than the interior rods, this tends to increase the conservative assumption made for the REBOL lattices of all rods having the same enrichment. This will offset the burnup gradient across any assemblies of consequence to the overall k-eff of the spent fuel storage array.

7.3 *Fuel Manufacturing Uncertainty*

The uncertainties due to the fuel manufacturing process include tolerance variations in enrichment, fuel pellet density, channel bulge, pellet diameter, clad diameter, pellet void volume, and gadolinia concentration. These independent uncertainty values have been statistically combined using the square root of the sum of the squares. The final combined fuel manufacturing uncertainty is calculated to be 0.00563 Δk_∞.

8.0 Conclusions

ATRIUM-10 fuel assemblies that meet the requirements defined in Table 2.1 can be represented as a single A10B-245L-0G0 REBOL lattice in subsequent spent fuel storage criticality safety analyses.

9.0 References

1. EMF-2808(P) Revision 0, *Criticality Safety Analysis for LaSalle Unit 2 Spent Fuel Storage Pool with Degraded Boraflex and Maximum ATRIUM-10 REBOL,* November 2002.

2. Exelon Design Analysis L-003241, Revision 0, "Criticality Safety Analysis for ATRIUM-10 Fuel LaSalle Unit 2 Spent Fuel Storage Pool (50% Degraded Boraflex Rack)", 11/21/2006. (AREVA archive # 38-9064454-000).

3. US Tool and Die Drawing 8601-7, Revision 4, "Comonwealth Edison Co. LaSalle County Station Unit-2 Spent Fuel Storage Racks Fuel Box Assembly & Groups", 10/24/1986. . (AREVA archive # 38-9064454-000).

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

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

6. 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.

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

Appendix A ATRIUM-10 Lattices Considered

The following lattices were considered as part of this evaluation.

LaSalle Unit 1			LaSalle Unit 2	
Lattice Identification	Cycle Loaded		Lattice Identification	Cycle Loaded
A10T-4306L-16G65	10		A10B-4503L-15G80	10
A10B-4507L-15G75	10		A10B-4511L-13G80	10
A10B-4504L-16G75	10		A10B-4326L-15G65	10
A10T-4305L-16G75	10		A10T-4313L-15G65	10
A10B-4510L-13G75	10		A10T-4302L-13G65	10
A10T-4307L-15G65	10		A10B-4494L-15G80	10
A10B-4504L-15G75	10		A10B-4502L-13G80	10
A10T-4040L-10G45	12		A10B-4253L-15G65	10
A10T-4042L-12GV80	12		A10T-4229L-15G65	10
A10B-3993L-12GV80	12		A10T-4021L-10G45	12
A10B-3618L-12G80	12		A10T-4022L-12GV80	12
A10T-4400L-10G45	12		A10B-3984L-12GV80	12
A10T-4451L-11G80	12		A10B-3726L-12G80	12
A10B-4459L-13GV80	12		A10T-4409L-10G45	12
A10B-4459L-12GV80	12		A10T-4455L-11G80	12
A10B-4466L-12G80	13		A10B-4481L-12GV80	12
A10B-4399L-12G65	13		A10T-2111L-0G0	10
A10T-3987L-12G65	13		A10B-1831L-0G0	10
A10T-3986L-12G40	13			
A10B-4454L-14G80	13			
A10T-4431L-14G80	13			
A10T-4444L-12G40	13			
A10T-3987L-12G80	13			

Appendix B Sample CASMO-4 Input

```
* DIM,10/
TTL * A10B-460L-11G60
TFU= 791.6
TMO= 560.3
VOI=00
FUE, 1,10.40239/ 2.7500
FUE, 2,10.40239/ 3.5700
FUE, 3,10.40239/ 4.1800
FUE, 4,10.40239/ 4.5900
FUE, 5,10.18405/ 4.6900,64016= 6.0000
FUE, 6,10.40239/ 4.7900
FUE, 7,10.40239/ 4.8900
FUE, 8,10.40239/ 4.9500
BWR,10,1.29540,13.40612,0.25400,0.66294,0.66294,1.2192,1
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
LPI
 1
 1 1
 1 1 1
 1 1 1 1
 1 1 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 1 1 1 1 1 1 1 1
 1 1 1 1 1 1 1 1 1 1
LFU
  1
  2  7
  4  5  8
  4  8  8  6
  4  8  8  8  0
  7  5  8  8  0  0
  4  8  8  8  0  0  0
  4  5  8  8  8  7  8  5
  2  7  5  8  5  8  6  6  3
  1  2  4  6  7  6  4  3  2  1
PUN 9*0 1
WRI  -20.5/'RES'
PDE,   52.3708, 'KWL'
DEP,0,.5,1,1.5,2,2.5,3,3.5,4,4.5,5,5.5,6,6.5,7,7.5,8,8.5,9,9.5,10,10.5,11,11.5,
   12,12.5,13,13.5,14,14.5,15,15.5,16,16.5,17,17.5,18,18.5,19,19.5,20,20.5
STA
TTL * in rack no boraflex  100c
RES,,12,12.5,13,13.5,14,14.5,15,15.5,16,16.5,17,17.5,18,18.5,19,19.5,20,20.5
TFU= 373.1
TMO= 373.1
VOI=00
BWR,10,1.29540,13.40612,0.25400,0.66294,0.66294,1.2192,1
PDE,0
MI1 7.92/347=100.0
FST 0.22860, 0.22860, 0.22860, 0.22860/
  0.095250, 0.095250, 0.095250, 0.095250/ 8*'MI1' / 8*'MOD' /
CNU,'FUE',54135,1.0E-14
STA
END
```

Distribution

Controlled Distribution

<u>Richland</u>

O. C. Brown
R. J. DeMartino
R. Fundak
R. E. Fowles
C. D. Manning
P. D. Wimpy