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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: Additional Information Supporting Request for License Amendment Regarding Spent Fuel Storage Pool Criticality

References:

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "License Amendment Request Regarding Spent Fuel Storage Pool Criticality," dated December 13, 2007

2. Letter from S. P. Sands (U. S. NRC) to C. G. Pardee (Exelon Generation Company, LLC), "LaSalle County Station, Units 1 and 2 – Request for Additional Information Related to Spent Fuel Pool Storage Requirements (TAC Nos. MD7900 and MD7901)," dated September 24, 2008

In Reference 1, Exelon Generation Company, LLC (EGC) requested 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 NRC requested additional information to support review of the license amendment request in Reference 2. The Attachments to this letter provide the requested information.

In Reference 1, EGC provided Holtec criticality analysis HI-2073758 as Attachment 3. Section 3.0 of the Holtec criticality analysis provided a list of the governing applicable codes, standards, and regulations. That section listed the 1974 version of ANSI/ANS-8.17, "Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors." That reference has been revised to the 1984 version of ANSI/ANS-8.17 for consistency with other criticality analyses that form the LSCS licensing bases. The criticality analyses documented in the Holtec report HI-2073758 are consistent with the 1984 version of ANSI/ANS-8.17.

EGC has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Attachment 1 of Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

Regulatory commitments are contained in Attachment 2 of 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 23rd day of October 2008.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Response to Request for Additional Information
2. Regulatory Commitments
3. Revised Markup of Proposed Technical Specifications Page

NRC Request 1

The LAR indicates that if a Boraflex panel reaches a certain amount of degradation, it can no longer be credited for reactivity control and the cell associated with that panel is declared "unusable." The above introduction states that the licensee's request to add the requirement for unusable storage cells to be filled with a "blocking device" to the TS to be an "apparently simple" request. "Apparently simple" because the LAR considers the "blocking device" to be an infallible means of preventing a fuel misloading into an unusable cell. However, the "blocking device" is not a permanent installation such as welding or locking a barrier in place. Rather, the "blocking device" is a movable object, as mobile as a fuel assembly (FA), and subject to essentially the same administrative controls as the FAs. The staff considers the misloading of a FA into an incorrect location in the SFP to be a creditable accident scenario. A recent event occurred at a boiling-water reactor (BWR) plant where two fuel assemblies were misloaded during a refueling outage. The LAR does not provide any justification for why misloading a FA into a cell instead of a "blocking device" is not a creditable event. Explain why misloading a FA into a cell instead of a "blocking" device' is not a creditable event.

Response

Section 9.1.1 of the Standard Review Plan (i.e., Reference 1) states that abnormal conditions should include consideration of fuel assemblies loaded into storage racks not approved for their storage. The fourth storage cell intended to be empty in the 3-of-4-analysis is a candidate for the misloaded bundle as it will be a cell not approved for fuel storage due to significant Boraflex degradation within a cell of the 3-of-4 array. It is reasonable to assume that absent an effective blocking device, there could be a misplacement of a spent fuel assembly, and this scenario is addressed in Section 7.8, Misloading of a Fuel Assembly in a Location Intended to be Empty, of Attachment 3 of Exelon Generation Company, LLC's (EGC's) license amendment request (i.e., Reference 2). Diverse, robust mechanical and administrative barriers provide the justification for why misloading a fuel assembly becomes a non-credible event. The stringent administrative controls governing the movement of fuel in conjunction with the tooling required to reposition blocking devices would preclude the event of misloading a bundle into a location that should have been empty of fuel. Therefore, the likelihood of this event occurring is so low the event becomes non-credible. However, as stated in Section 7.8, calculations were performed for a 6x6 array of storage cells with the central blocking device replaced with a fuel assembly. Table 7.5 of Attachment 3 of Reference 2 contains the results and demonstrates 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. It should be further noted that this result is substantially conservative in that it reflects all fuel being modeled in the x-y-z directions at its most reactive condition; whereas, due to axial burnup profiles, such a reactivity condition will never occur in the spent fuel pool.

EGC plans to implement a series of actions to prevent placing a fuel assembly into a cell intended to have a blocking device. In order to confirm that 1-of-4 fuel storage cells have been physically restricted from containing a fuel bundle, a mechanical blocking device will be used to block the fourth cell. EGC does not intend to routinely remove these devices once installed, but might need to do so from time to time. Prior to removing a blocking device from a cell, EGC will remove fuel to ensure that no more than two bundles remain in the affected 2x2 arrays to ensure that a single misloaded bundle will not create a condition not already analyzed by the 3-of-4 analysis.

EGC proposes to use a specifically designed blocking device for the fourth cell, rather than a fuel channel, double blade guide, or other typical spent fuel pool item. The blocking device design will be approximately the same height as the cell top and extend the length of the cell to the rack base. The device top is clearly distinguishable from a bundle handle or blade guide, and cannot be engaged for lifting by the same grapple as used to engage and lift the bail handle of the fuel bundles or blade guides. Rather, its upper region will have a lifting eye to be engaged by a J-hook or similar hook/handle tool. This is fundamentally different in both appearance and tooling, which will prevent mistakenly selecting and moving the blocking device. Strict administrative procedures govern the use of a J-hook. The J-hook is not intended for everyday use or for the movement of fuel assemblies. Because of this, the proposed blocking device is a robust physical barrier to preventing misloading of a fuel assembly. Inadvertent removal of the blocking device with the refuel bridge is precluded since the blocking device will not have a bail handle and can only be moved with a J-hook. Robust administrative processes and procedures govern the use of a J-hook and provide a second barrier to inadvertent removal of a blocking device. This design feature is consistent with the Kopp letter guidance (i.e., Reference 3) such that the J-hook prevents the blocking device from being removed either inadvertently, or intentionally without unusual effort such as the necessity for special equipment maintained under positive administrative control.

In addition to the mechanical barrier, several administrative controls and processes also preclude a misloading event. First, using the Boraflex surveillance program, EGC identifies cells that are Boraflex depleted and require a blocking device within the 2x2 array. This process is described as follows:

- Project the depletion of the Boraflex panels using the RACKLIFE computer code.

- Identify cells that will become depleted over the next projection period. A cell is depleted if one or more panels are depleted or if the cell average exceeds the acceptance criteria of 57.5% depletion. Also, one depleted panel can impact two cells, depending on cell location. A panel face adjacent to a location not designed for fuel storage (e.g., blade guide racks, pool walls, open water) is not considered depleted as it does not contain any Boraflex, by design, as the neutron leakage in these positions is high.

- Identify boundary cells. Boundary cells are cells that are not degraded but are a king's move away (i.e., within one cell horizontally, vertically, or diagonally) from a degraded cell.

- Select the locations for the blocking devices such that every degraded cell and every boundary cell is either a king's move from a blocking device or contains a blocking device.

The cells identified as degraded and the cells requiring blocking devices would be documented in a design change package (DCP) under the EGC configuration control process. The DCP would include document changes to affected plant drawings, procedures, and databases reflecting the selected locations of the blocking devices and degraded cells. Move sheets would direct the movement of the blocking devices and installation would be controlled by the work control process. Move sheets, prepared and reviewed by qualified reactor engineers, would control the placement of the blocking devices within the 2x2 array containing the depleted cell. These move sheets constitute both a written procedure and a plan that designates initial and final locations. A fuel movement spotter provides independent verification of each fuel move. This consists of independent verification of proper bridge location for both picking up and storing of fuel assemblies and blocking devices. The SHUFFLEWORKS database would also be updated to reflect the locations identified as depleted and appropriate locations blocked within the 2x2 arrays of cells. This would prevent the move sheet builder software from allowing fuel to be placed in these locations. In addition, site procedures governing fuel handling in the spent fuel pool (SFP) would be revised to list the depleted locations and those locations containing blocking devices. Additional information regarding the process used to control movement of items within the SFP is provided in response to NRC Request 27.

NRC Request 2

The LAR Attachment 1 states, "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." However, both the Holtec criticality analysis in the LAR Attachment 3 (HI-2073758) and the AREVA NP, Inc. (ANP) criticality analysis in the LAR Attachment 4 (ANP-2684) indicate that, in the absence of Boraflex, the maximum reactivity occurs at higher temperatures. Explain this apparent contradiction.

Response

The statement in paragraph 5 in Section 3 of Attachment 1 to Reference 2 that "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," applies to the configuration with the neutron absorber (i.e., Boraflex) present in the SFP. With the neutron absorbing material present, lower temperatures (i.e., 4°C) provide a more reactive condition. This has been demonstrated through previous SFP criticality calculations for both the LaSalle County Station (LSCS) Unit 2 pool (i.e., consisting of the Boraflex neutron absorbing material) and the LSCS Unit 1 pool (i.e., consisting of the Boral neutron absorbing material). However, absent the SFP neutron absorbing material, higher temperatures provide a more reactive condition. Without Boraflex, the 3-of-4 loading configuration uses water and steel as the dominant neutron absorbing material. Because water density decreases with increasing temperature, the level of neutron capture in the water will also decrease with increasing temperature. This leads to a higher k_{eff} result at 100°C for the no Boraflex condition.

The criticality analysis performed by Holtec, as documented in Attachment 3 of Reference 2, was performed at the water temperature and density that corresponds to the maximum reactivity. Section 7.2 and Table 7.3 of Attachment 3 of Reference 2 demonstrate that the maximum reactivity is associated with the maximum temperature of 254°F. Since the SFP is an open configuration, a temperature of this magnitude would be associated with an accident (i.e., not a normal) condition. Because the MCNP-4A calculations are valid at, and were performed at, a temperature of 300K (80.33°F, 27°C), the difference in reactivity between 27°C and 123°C is applied as a temperature bias in the calculation of the maximum k_{eff} as shown in Table 7.1 of Attachment 3 of Reference 2.

NRC Request 3

The Technical Analysis in the LAR Attachment 1 states, "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. Exelon has evaluated this lattice and determined that it is bounded by the 2.45 wt percent U-235 uniform enriched ATRIUM-10 no Gadolinium lattice modeled in the criticality analysis." This appears to be justified by a table showing the in-core k_∞ of limiting lattices. As described, the values in the table appear to have been calculated in standard cold core geometry (SCCG). The table shows the 2.45 wt percent U-235 uniform enriched ATRIUM-10 no Gadolinium lattice maximum SCCG k_∞ exceeding that of the SCCG k_∞ attributed to the limiting legacy FA.

However, ANP-2684 (LAR Attachment 4) Table 6.1 shows the 2.45 wt percent U-235 uniform enriched ATRIUM-10 no Gadolinium lattice maximum in-rack k_∞ to be at least 0.0135 Δk lower than the maximum Unit 1 Cycle 13 in-rack k_∞.

a) Since the in-rack k_∞ for the 2.45 wt percent U-235 uniform enriched ATRIUM-10, and no Gadolinium lattice is lower than the maximum Unit 1 Cycle 13 in-rack k_∞, how can the 2.45 wt percent U-235 uniform enriched ATRIUM-10 no Gadolinium lattice be limiting?

b) How do the in-core reactivity calculations translate to in-rack reactivity?

Response

The in-rack reactivity of the ATRIUM-10 2.45 wt% reactivity equivalent at beginning of life (REBOL) lattice does not bound the in-rack reactivity of all ATRIUM-10 lattices used at the LSCS reactors. This is demonstrated in Table 6.1 of ANP-2684 where those lattice designs that are bolded have higher in-rack reactivity than an ATRIUM-10 2.45 wt% REBOL lattice. These bolded designs that have higher in-rack reactivity are near the top of the ATRIUM-10 assemblies (i.e., just below the natural U top blanket) and comprise a maximum of 12 inches in total length. However, in-rack calculations for a simple fuel assembly model comprised only of the 2.45 wt% REBOL lattice and geometry type are bounding of in-rack calculations for actual ATRIUM-10 fuel bundle designs that may contain one of these higher reactivity designs near the top, as explained below and documented in ANP-2684 Table 6.2.

As stated in the previous paragraph, Table 6.1 of ANP-2684 identifies maximum CASMO-4 in-rack k_∞ reactivity, at any burnup level, for the ATRIUM-10 fuel lattices that have been built for LSCS Units 1 and 2 that are more reactive (i.e., in the in-rack configuration) than the 2.45 wt% REBOL lattice. These high reactivity lattices are characteristic of the lattices that will be loaded in specific axial locations (i.e., near the top of the assembly) in LSCS ATRIUM-10 fuel assemblies and can be represented (i.e., bounded) by a single enrichment 2.70 wt% REBOL lattice as shown in the following table. Note that the REBOL lattices are defined with higher k_∞ values than the lattices they represent.

Zone Description for as-fabricated assemblies	Zone Length for as-fabricated assemblies (inches)	As-fabricated number of fuel rods in zone	ANP-2684 Table 6.1 lattices used in as-fabricated assemblies (Case #)	ANP-2684 Table 6.1 REBOL Lattice bounding of as-fabricated assemblies (Case #)
Top Natural Blanket	5 or 11	83	--	--
Top Enriched	12	83	7 – 11	14
Middle Enriched	36 or 30	83	5 – 6	13
Bottom Enriched	90	91	1 – 4	12
Bottom Natural Blanket	6	83	--	--

The primary purpose of ANP-2684 is to define a single lattice assembly model that can be conservatively used in criticality codes/analyses (e.g., KENO or MCNP) to represent the ATRIUM-10 assemblies that have been built for the LSCS Units. The KENO V.a in-rack reactivity comparison provided in Table 6.2 shows that a bottom lattice at 2.45 wt% U-235 modeled for 149 inches (i.e., Case 1 with KENO V.a k_{eff} of 0.9165) is more reactive than an assembly model that uses five axial enrichment/geometry zones that are bounding of as-fabricated assemblies (i.e., Case 2 – consistent with the last column in the above table, with KENO V.a k_{eff} of 0.9149). More detail about this KENO comparison is provided in the response to NRC Request 25.

Analyses have demonstrated that lattices with similar in-core reactivity but from different fuel product lines may indeed not show similar reactivity in the in-rack configuration. As stated in the NRC Request, the table on page 5 of 12 of Attachment 1 of the LAR does show a comparison of the maximum in-core k_∞ for each of the three product lines that have previously been used at LSCS (i.e., legacy GE 8x8 fuel, ATRIUM-9B fuel, and GE14 fuel). The table also shows the maximum in-core reactivity of the ATRIUM-10 fuel assembly design at LSCS, with the in-core reactivity of the ATRIUM-10 2.45 wt% lattice shown as slightly higher. The purpose of this table is to demonstrate that the ATRIUM-10 fuel assembly product line is more reactive than the other LSCS assembly product line types.

However, to quantitatively confirm that this in-core reactivity difference between the ATRIUM-10 design and the legacy fuel designs will translate to the ATRIUM-10 design being limiting in an in-rack configuration, in-rack CASMO-4 calculations were performed for the most reactive fuel lattice of each assembly product line type that has been used in the LSCS reactors. For each assembly product line type, the more reactive lattice designs were identified using a comparison of the Gadolinia concentration and U-235 enrichment levels. CASMO-4 in-rack calculations were then performed for the most reactive lattices with the limiting lattice results shown in the table below. The lattices used in the 12" fuel zone beneath the top of the ATRIUM-10 assembly are not included in this comparison because they are limited to 12" in length in ANP-2684. Additional detail regarding this 12" fuel zone, and how its higher reactivity levels are acceptable, is provided above, and in response to NRC Request 25. The table below lists the highest in-rack k_∞ result for each LSCS fuel product line and the temperature at which it was obtained. Also included for completeness are in-rack results for ATRIUM 10XM lead use assemblies, since eight assemblies are scheduled to be loaded during the Spring 2009 refueling outage. The results in the table below confirm that the primary ATRIUM-10 bounding lattice (i.e., A10B-460L11G60) is more reactive than the limiting legacy fuel product line lattices and that the ATRIUM-10 REBOL lattice is the most reactive of all. There are no biases or uncertainties applicable to this comparison of limiting lattices.

Fuel Assembly Product Line Type	**Description of Limiting Lattice (Enrichment and Gadolinia Content)**	**CASMO-4 In-Rack k_∞**
		No Boraflex (100°C)
GE 8x8	340L-7G30	1.0827
ATRIUM-9B	458L-8G60	1.0973
GE14	435L-6G70-9G60	1.0492
ATRIUM 10XM	4056L-12G40	1.0949
ATRIUM-10	460L-11G60	1.0981*
ATRIUM-10	REBOL 245L-0G0	1.1069*

*In-rack k_∞ results included in Table 6.1 of ANP-2684.

NRC Request 4

Based on the following excerpts from the LAR Attachment 1, the licensee's proposed identification and control of the 3-of-4 storage configuration appears to be inadequate. From the Interfaces Between Areas of 3-of-4 and 4-of-4 Storage section of the Technical Analysis of the LAR Attachment 1, the following controls are proposed to ensure the criticality analysis remains valid.

a) "Each cluster of four storage cells (i.e., 2x2) must meet either the criteria for 4-of-4 storage or the criteria for 3-of-4 storage."

b) "In each cluster of four storage cells (i.e., 2x2), 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."

It is unclear how these controls will ensure the criticality analysis remains valid, as they could be satisfied with an arrangement that would leave a fuel assembly in the cell with the actual degraded Boraflex panels while having the empty cell be some other cell. This would allow the storage cells with the degraded Boraflex panels to be part of a 4-of-4 storage configuration with other fuel assemblies, a scenario the licensee has indicated does not meet their TS or 10 CFR 50.68. Explain why that would be acceptable.

Response

Since the criticality analysis assumes no Boraflex is present in any of the cells within the 2x2 array, as long as one of the four cells in every possible 2x2 set of cells that has a degraded cell contains a blocking device, the conditions of the criticality analysis are met. To satisfy the conditions of the criticality analysis, it is not necessary that the blocking device be inserted in the degraded cell itself, or that the degraded cell not contain a fuel assembly. However, insertion of the blocking device into the degraded cell would be prudent to minimize locations to be blocked. Additional information is provided in response to NRC Request 5. Figure 1 in the response to NRC Request 5 demonstrates how a degraded cell may contain a fuel assembly, while a non-degraded cell within the 2x2 array may contain a blocking device. The 3-of-4 criticality criteria are met by the configurations shown in Figure 1 in the response to NRC Request 5. To simplify implementation of the 3-of-4 criticality analysis, EGC will commit to use of the blocking device in a periodic repetitive pattern, as discussed in response to NRC Request 6.

NRC Request 5

It is not clear that all of the affected 2x2 arrays are being identified. A storage cell is not part of just one 2x2 array. Unless it is on the periphery, each storage cell is at the center of a 3x3 array which comprises four different 2x2 arrays. Additionally, from the description in the LAR, it appears that the Boraflex panels are shared by two cells, meaning the degradation of one panel actually affects two cells, creating a 4-by-3 array of cells with six separate 2x2 arrays. It is not clear whether the adjacent cell is also being identified as 'unusable' and its associated 2x2 arrays being controlled. Provide a description of how the affected 2x2 arrays are being identified and controlled so that the staff may make a reasonable assurance decision that all of the appropriate affected 2x2 arrays are being included.

Response

A single degraded storage panel causes two degraded storage cells, which are then part of six 2x2 arrays. Each of these 2x2 arrays must satisfy the acceptance criteria for either 3-of-4 or 4-of-4 storage. There is no need to consider the 3x3 array with the degraded cell in the center as satisfying the conditions for each 2x2 set of cells that contains a degraded cell is sufficient.

Therefore, in each of the six 2x2 arrays that can be considered containing a degraded storage cell, there must be a blocking device in one of those four cells. In such a configuration, a minimum of two blocking devices would be required (i.e., most likely placed face adjacent to the degraded panel but not necessarily placed face to face as shown below) to satisfy the 3-of-4 storage criteria for each of the six 2x2 arrays created by the degraded panel.

Figure 1 provides several other configurations of degraded storage cells and the requirements for blocking devices. In the figure below, panels that exhibit an unacceptable level of Boraflex are designated by a dashed line. Boraflex panels that do not exhibit an unacceptable level of

Boraflex are designated by a solid black line. The first letter in each cell specifies whether a storage cell is considered degraded (D) or not degraded (N). A storage cell is considered degraded if one or more of the four surrounding Boraflex panels are considered degraded to an unacceptable level. The second letter in each cell specifies whether the storage cell contains a blocking device (B) or is filled with spent fuel (F).

DB	Degraded Cell, Blocking Device
DF	Degraded Cell, Filled
NB	Non-Degraded Cell, Blocking Device
NF	Non-Degraded Cell, Filled

Degraded Panel

Non-Degraded Panel

NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF
NF	NF	DB	NF	NF	NF	DB	NF	NF	NF	DB	NF	NF	NF	NF
NF	NF	DF	NF	NF	NF	DB	NF	NF	NF	DF	NF	NF	NF	NF
NF	NF	DB	NF	NF	NF	NF	NF	NF	NF	NB	NF	NF	NF	NF
NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF
NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NB	NF	NF	NF	NF
NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	DF	NF	NF	NF	NF
NF	NF	NF	NB	NF	NF	NF	NF	NB	DF	DB	DF	NB	NF	NF
NF	NF	NF	DF	NF	NF	NF	NF	NF	NF	DF	NF	NF	NF	NF
NF	NF	NF	DB	DF	NB	NF	NF	NF	NF	NB	NF	NF	NF	NF
NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF
NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF	NF

Figure 1: Examples of Implementing the 3-of-4 and 4-of-4 Interface Configurations

Degraded Boraflex panels are identified using RACKLIFE projections. The SHUFFLEWORKS database will also be updated to reflect the locations identified as depleted and those locations blocked with blocking devices.

To simplify implementation of the 3-of-4 criticality analysis, EGC will commit to use of the blocking device in a periodic repetitive pattern, as discussed below in response to NRC Request 6. This commitment would be such that all blocking devices would be installed in every other cell in every other row of the spent fuel storage pool sufficient to cover the depleted cells in a 3-of-4 configuration (i.e., the blocking devices would be installed in a pattern that mimics Figure 7.1 of HI-2073758). The coverage would be extended sufficiently in the x and y directions to cover the depleted cells of interest such that the 3-of-4 configuration requirements are met. This would preclude blocking devices being placed as shown in Figure 1. Figure 2 provides an example that depicts blocking devices being installed in a periodic repetitive pattern.

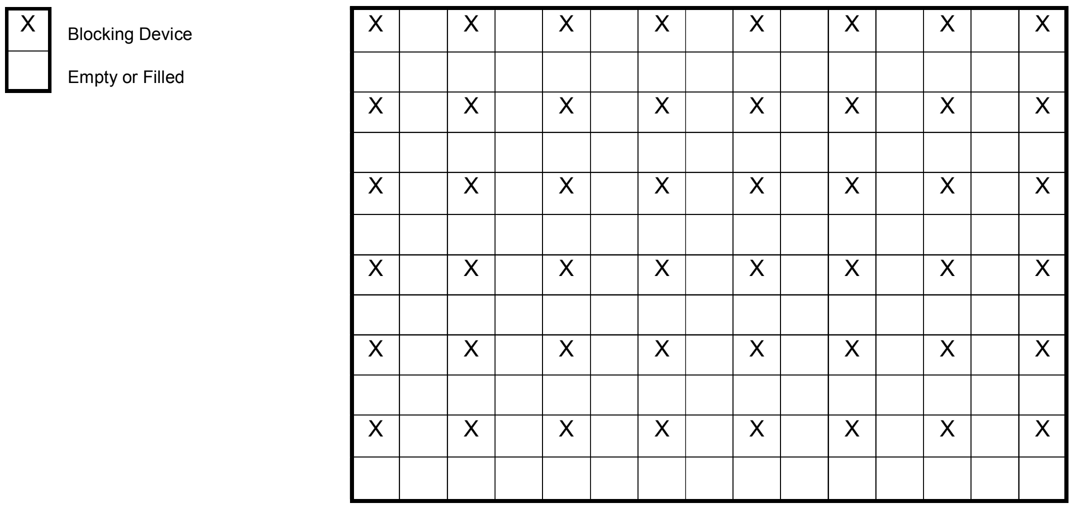

Figure 2: Periodic/Repetitive Implementation of Blocking Devices for the 3-of-4 Criticality Analysis

<u>NRC Request 6</u>

The licensee is proposing the following requirement be added to TS 4.3.1.1 which states, "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." According to the LAR, a cell that has even one degraded Boraflex panel, cannot maintain the requirements of 4.3.1.1.a. Eventually all SFP storage cells will have at least one degraded Boraflex panel. This proposed TS would require those cells to have blocking devices. This requirement would prevent the removal of the blocking device from a storage cell so that the storage cell could hold a FA. Explain how the 3-of-4 storage configuration is compatible with the proposed TS requirement.

<u>Response</u>

The requirement should be reworded to say "For Unit 2 only, for any 2x2 arrangement of storage cells for which the Boraflex neutron absorber cannot maintain the requirements of 4.3.1.1.a, a blocking device shall be installed in one of the four cells of the affected 2x2 array." A revised markup of the affected Technical Specifications page is provided in Attachment 3.

Although EGC does not intend to routinely remove the blocking devices once installed, there may be a need to do so from time to time. Prior to removing a blocking device from a cell, EGC will remove fuel to ensure that no more than two bundles remain in the affected 2x2 arrays to ensure that a single misloaded bundle will not create a condition not already analyzed by the 3-of-4 analysis.

EGC's responses to NRC Requests 4, 5, and 6 are based upon the configurations supported by the criticality analyses of HI-2073758. Those configurations, as specifically demonstrated in the information contained in the responses to NRC Requests 4 and 5, allow for blocking devices to be placed in "non-periodic/non-repetitive patterns" and still meet the criticality compliance criteria of the Holtec 3-of-4 analyses. EGC recognizes the complexity in the administrative controls and processes that these types of configurations would present. Therefore, to minimize such complexity in the administrative controls and processes, and ensure fuel mislocation remains a non-credible event, EGC will commit to using the blocking device in a periodic repetitive pattern. This commitment would be such that all blocking devices would be installed in every other cell in every other row of the spent fuel storage pool sufficient to cover the depleted cells in a 3-of-4 configuration (i.e., the blocking devices would be installed in a pattern that mimics Figure 7.1 of HI-2073758). The coverage would be extended sufficiently in the x and y directions to cover the depleted cells of interest such that the 3-of-4 configuration requirements are met. This would preclude blocking devices being installed in face-to-face configurations as shown in Figure 1 in the response to NRC Request 5.

NRC Request 7

Provide a rationale as to why none of the interface requirements or definitions are being placed into the LSCS TS. 10 CFR 50.36(d)(4) which states, "Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section." Provide a revised TS proposal that includes these items in the LSCS Unit 2 TS.

Response

Section 4.3.1.1 of the Technical Specifications provides the criteria for criticality compliance that k_{eff} is less than or equal to 0.95 for the SFP if fully flooded with unborated water. The Updated Final Safety Analysis Report (UFSAR) contains the specific requirements for non-degraded SFP Boraflex criticality compliance, including geometric arrangements and references to criticality analysis documentation, to satisfy these criteria for all fuel product lines used at LSCS. The UFSAR will be updated to include the interface requirements and criticality analysis documentation associated with the 3-of-4 criticality licensing basis for the degraded spent fuel Boraflex conditions.

NRC Request 8

In the No Significant Hazards Consideration (NSHC) section of the Regulatory Analysis of the LAR Attachment 1, it indicates there is no possibility of a new or different kind of an accident from any accident previously evaluated stating, "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." The NRC staff disagrees with this conclusion. The NRC staff does not view the use of typical SFP items such as "…e.g., fuel channel, blade guide, etc.…" as blocking devices and "…controls for movement of a blocking device that are similar to the controls that govern fuel movement…" to be sufficiently robust to preclude the misloading of a FA since misloadings do occur despite the controls that govern fuel movement. Additionally, the process as described in the LAR is dynamic in that it appears likely that the empty cell in the 3-of-4 storage configuration will change over time as different Boraflex panels are identified as degraded. Labeling cells as "unusable" when in fact they may be used in the 3-of-4 storage configuration is a misnomer and may lead to confusion with respect to the actual use of the cell and increase the possibility of a misloading event. Therefore, the NRC staff considers the misloading of a FA into a cell that is unusable as a creditable event and as an accident different from those previously evaluated at LSCS. Provide a revised NSHC that includes a misloaded fuel assembly as an accident not previously analyzed at LSCS.

Response

The evaluation of an assembly mislocation as part of a criticality analysis is not a new event and is an accident previously evaluated for LSCS criticality analyses. For these past analyses, the assembly mislocation occurs only as a misplacement outside of the fuel racks (as all locations within the fuel racks are considered occupied). The response to NRC Request 1 provides information that supports the conclusion that the mislocation event is not credible. As stated in the response to NRC Request 1, Table 7.5 of Attachment 3 of Reference 2 contains the results and demonstrates 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. It should be further noted that this result is substantially conservative in that it reflects all fuel being modeled in the x-y-z directions at its most reactive condition and that all fuel is modeled as the limiting ATRIUM-10 assembly. Due to the lower reactivity of legacy fuel and axial burnup profiles, such a reactivity condition will never occur in the SFP.

Fuel channels, blade guides, etc. will not be used as blocking devices. Only the specifically designed blocking devices described in the response to NRC Request 1 will be used.

The labeling of cells as "unusable" has been removed to eliminate the possibility of confusion. Cells are labeled as degraded or non-degraded to be consistent with the criteria of degradation as identified in the 4-of-4 and 3-of-4 analyses. Since the criticality analysis assumes no Boraflex is present in any of the cells within the 2x2 array, as long as one of the four cells in each 2x2 set of cells that has a degraded cell contains a blocking device, the conditions of the criticality analysis are met. To satisfy the conditions of the criticality analysis, it is not necessary that the blocking device be inserted in the degraded cell itself or that the degraded cell not contain a fuel assembly.

To ensure fuel mislocation remains a non-credible event, EGC will commit to using the blocking device in a periodic repetitive pattern. This commitment would be such that blocking devices would be installed in every other cell in every other row of the spent fuel storage pool sufficient to cover the depleted cells in a 3-of-4 configuration (i.e., the blocking devices would be installed in a pattern that mimics Figure 7.1 of HI-2073758).

NRC Request 9

In the Section 2.1, Code Validation, the Holtec criticality analysis states, "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." References 3 and 4 are not provided nor are they publicly available. There is no generic Topical Report for CASMO-4, for either in-core analyses or in-rack analyses. All current approvals for using CASMO-4 are based on a site specific acceptance. There was no site specific justification for using CASMO-4 provided. Therefore, the staff needs the following:

a) LCSC site specific justification for using CASMO-4

b) No code validation for the use of CASMO-4 was provided. It does not appear that CASMO-4 code bias and uncertainty were determined or applied. Provide these in accordance with references 2 and 3.

<u>Response</u>

CASMO-4 is not used in this application to calculate absolute reactivities, but is only used to determine relative reactivity differences for temperature variation and manufacturing tolerances. References 3 and 4 of Attachment 3 of Reference 2 are Studsvik proprietary documents related to the appropriateness of CASMO-4 for calculating the multiplication factor, k_{eff}. References 3 and 4 of Attachment 3 of Reference 2 were previously provided to the NRC in support of approval of EMF-2158, as discussed in response to NRC Request 19, as documented in Reference 4.

Holtec replaced the CASMO-3 code with the CASMO-4 code in approximately mid-1999 for calculating the reactivity effects of manufacturing tolerances, moderator temperature, and depletion effects. CASMO-4 has been previously used and approved by the NRC over the past ten years on multiple licensing efforts by Holtec for spent fuel storage racks. Specifically, CASMO-4 has been reviewed and approved for use on the following spent fuel pool analyses for calculating the reactivity effect of moderator temperature variation and manufacturing tolerances.

Pressurized Water Reactors (PWRs)	Boiling Water Reactors (BWRs)
Crystal River 3	Clinton
Arkansas Nuclear 1 & 2	Nine Mile Point Unit 2
Harris	Cooper
St. Lucie	Fermi
Diablo Canyon	Harris (Brunswick BWR fuel in Harris PWR spent fuel pool)
Turkey Point	
V.C. Summer	
Three Mile Island	
Comanche Peak	
Davis-Besse	
Robinson	
Sequoyah	

From the above list of plants, the following specific subset of NRC issued SERs and amendment approval references are identified where CASMO-4 and MCNP4a have been used by Holtec for spent fuel pool criticality analyses.

- Letter from F. E. Saba (U. S. NRC) to J. S. Forbes (Entergy Operations, Inc.), "Arkansas Nuclear One, Unit No. 1 – Issuance of Amendment for Use of METAMIC® Poison Insert Assemblies in the Spent Fuel Pool (TAC No. MD2674)," dated January 26, 2007

- Letter from K. N. Jabbour (U. S. NRC) to C. M. Crane (AmerGen Energy Company, LLC), "Clinton Power Station, Unit 1 – Issuance of an Amendment – RE: Onsite Spent Fuel Storage Expansion (TAC No. MC4202)," dated October 31, 2005

- Letter from S. N. Bailey (U. S. NRC) to D. E. Young (Crystal River Nuclear Plant), "Crystal River, Unit 3 – Issuance of Amendment Regarding Fuel Storage Patterns in the Spent Fuel Pool (TAC No. MD3308)," dated October 25, 2007

The use of CASMO-4 by Holtec for SFP licensing activities on these BWR plants, and NRC approval of that use, provides the justification for using CASMO-4 for relative reactivity calculations for the LSCS SFP 3-of-4 criticality analysis.

Holtec has validated the use of CASMO-4 within their NRC approved Quality Assurance (QA) Program as documented in Holtec Report HI-981945, "QA Documentation and Validation of CASMO-4." This validation is governed by the Holtec QA Program, which is subjected to NRC audit every three years.

Code biases and uncertainties are applied in determination of an absolute k_{eff} value. This determination is performed with the MCNP 4a code and documented in the Holtec analyses (see the response to NRC Request 12). No code bias or uncertainty for CASMO-4 is necessary to be applied in this licensing submittal as CASMO-4 is used only for differential reactivity calculations. Additionally, the CASMO-4 calculations are performed in a conservative manner (i.e., single storage cell, an infinite array of storage cells) and applied to the 3-of-4 configuration, such that the calculated reactivity effect bounds the actual reactivity effect associated with manufacturing tolerances and temperature variation.

NRC Request 10

In the Section 4.0, Assumptions, the Holtec criticality analysis states, "To assure that the true reactivity will always be less than the calculated reactivity, the following conservative assumptions were made:"

One assumption is that, "Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water." This appears to be incongruent with the subsequent statement in a later section that, "Therefore, MCNP-4A calculations were performed to verify that including the channel in the final analysis is conservative." Both the conclusion that it is conservative to model the FA channeled and the assumption that it is conservative to not model the FA grids and end fittings appear to be balancing the absorption of structural components against increasing the amount of over moderation. Explain why both assumptions are conservative at the same time.

Response

The presence of channels and grid spacers in the active fuel region of the assembly has the effect of changing the water-to-fuel ratio. The BWR channels extend the entire length of the fuel assembly and are located around the outside of the assembly, between the fuel assembly and the storage cell walls. The major effect of the Zircaloy fuel channel is in displacing the water between fuel assemblies. The response to NRC Request 14 quantifies the reactivity decrease associated with removal of the fuel channel. Grid spacers are located within the assembly, between adjacent fuel pins. Additionally, they are located at periodic intervals along the axial length of the fuel assembly. The difference in the location, either within the assembly or without, of these types of hardware dictates their effect on reactivity.

To confirm that neglecting the grid spacers is conservative, an MCNP4a calculation was performed which includes the grid spacers. The eight grid spacers are modeled as solid zirconium, 1.2 inches in height, spaced equidistant from one another along the axial length of the active fuel region. The actual grid spacers are constructed of a Zircaloy-4 structure, with Inconel springs. The Inconel springs are not credited in the analysis. The table below presents the results of the calculated reactivity and compares it to the reference case.

Case	Calculated k_{eff}	Sigma
Reference	0.9261*	0.0007
Spacers Included	0.9172	0.0006

* Result included in Table 7.1 of HI-2073758

The performed analysis confirms that neglecting the grid spacers in the rack calculations is conservative as the Reference case results in a greater reactivity than the case with the spacers included.

NRC Request 11

In Section 5.1, Fuel Assembly Specifications, of the Holtec criticality analysis, the vendor has used the ATRIUM-10 FA with a reactivity equivalent uniform enrichment of 2.45 w/o 235U as the limiting assembly. This is what is referred to as the reactivity equivalent fresh fuel enrichment (REFFE). The REFFE is intended to represent the maximum reactivity state. BWR fuel typically has higher enrichments than 2.45 w/o, but they also typically have Gadolinium burnable absorber included. As the Gadolinium depletes with burnup, the reactivity increases to a maximum. The REFFE equates the maximum reactivity of the Gadolinium depleted fuel with a low enriched fresh FA. Care must be used with the REFFE as changes in the model, such as using a REFFE determined in a 4-of-4 storage configuration in a 3-of-4 storage configuration, can create non-conservative results. NUREG/CR-6683, A Critical Review of the Practice of Equating the Reactivity of Spent Fuel to Fresh Fuel in Burnup Credit Criticality Safety Analysis for pressurized-water reactor (PWR) Spent Fuel Pool Storage, (Reference 5) addresses these concerns. The REFFE was provided by the licensee. The ANP criticality analysis in LAR Attachment 4 is the justification for the 2.45 wt percent U-235 uniform enriched ATRIUM-10 no Gadolinium lattice REFFE. It is not clear that these issues have been adequately addressed. Describe how this potential non-conservatism has been avoided.

Response

EGC agrees that care must be taken in using the REFFE or REBOL approach. The 2.45 w/o ATRIUM-10 REBOL design has been determined for an in-rack configuration with no Boraflex present. This in-rack configuration with no Boraflex present was also the basis of the relative reactivity comparisons with legacy fuel product lines used at LSCS (see the table in the response to NRC Request 3). The 3-of-4 criticality analyses of HI-2073758 are also performed with this same basis (i.e., an in-rack configuration with no Boraflex present).

Section 4.3.2 of NUREG/CR-6683 (i.e., Reference 5) addresses the 3-of-4 loading configuration and indicates that the REFFE in a 4-of-4 configuration provides conservative results when it is loaded in a 3-of-4 configuration using an empty fourth storage location. As excerpted from Section 4.3.2 of NUREG/CR-6683, "When a REFFE assembly is placed in storage with an empty cell (or a less reactive assembly), the REFFE approach yields conservative results." Additionally when AREVA defines the REBOL lattice for BWR storage analyses, a conservative factor of at least 0.005 Δk is typically maintained to account for code biases and model uncertainties.

NRC Request 12

In Section 7.1, Manufacturing Tolerances, of the Holtec criticality analysis discusses the uncertainties associated with manufacturing tolerances associated with the fuel assemblies and storage racks. The section states, in part, "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." While this is the appropriate application of the tolerances to determine the reactivity uncertainty associated with the tolerances, this description is only applied to two of the dozen or so manufacturing tolerances which affect reactivity. Not all of the manufacturing tolerances which affect reactivity are discussed. The other tolerances, which are discussed, are lumped together in a single value in Table 7.2 with no discussion of where it came from, how it was derived, or its basis. Provide the justification and basis for this value. Include sufficient detail for the staff to independently reach a reasonable assurance determination regarding its use.

Response

The reactivity uncertainty effect of fuel manufacturing and fuel storage rack tolerances has been readdressed to clarify the tolerance reactivity uncertainties that were included in Table 7.1 of HI-2073758. The table below provides all of the uncertainties considered as part of the Rack and Fuel Tolerance reactivity value of 0.0111 in Table 7.1 of HI-2073758. The information below demonstrates the conservative nature of the Rack and Fuel Tolerance reactivity component of Table 7.1 of HI-2073758.

Reactivity Uncertainty of Fuel Tolerances Rack and Fuel Uncertainty	
Parameter	**Δk_∞**
Fuel Enrichment	0.00211
Fuel Density	0.00100
Channel Bulge	0.00383
Pellet Diameter	0.00017
Clad Diameter	0.00116
Pellet Void Volume	0.00027
Gadolinia	0.00318
Fuel Rod Pitch	0.0007

Reactivity Uncertainty of Rack Tolerances	
Parameter	**Δk_∞**
Storage Cell Inner Dimension	0.0008
Wall Thickness	0.0084
Storage Cell Pitch	0.0009

Reactivity Uncertainty Summary	
Parameter	**Δk_∞**
Statistically Combined Value	0.0102
Reported Uncertainty in Table 7.1 of HI-2073758	0.0111

The uncertainty value of 0.0111 Δk_∞ reported in HI-2073758 bounds the statistically combined uncertainty value of 0.0102 Δk_∞.

<u>NRC Request 13</u>

The Holtec criticality analysis has no discussion about the burnup (BU) uncertainty. This is reasonable for the Holtec analysis as the REFFE is given to them by the licensee as an input and the REFFE is fresh fuel. The BU uncertainty should be included in determining the REFFE. But there is no discussion regarding BU in the ANP criticality analysis in LAR Attachment 4.

It is not clear what uncertainties are included in the ANP criticality analysis in LAR Attachment 4. Provide the uncertainties that are included in the ANP criticality analysis.

Response

No uncertainties are included in the AREVA documentation because the purpose of ANP-2684 is to present reactivity comparison calculations that support the limiting REBOL lattice used in the Holtec report HI-2073758. In ANP-2684, in-rack reactivity calculations for a bundle composed of the 2.45 wt% design are compared to in-rack reactivity calculations for a bundle consisting of a more reactive design located in a 12" zone length below the top natural U blanket. In-rack reactivity comparison results are presented in the response to NRC Request 3. When comparing k_∞ values in this manner, the uncertainties are very small because code biases are being applied to all calculations in a similar manner. The fuel and manufacturing uncertainty values that are reflected in the maximum k_{eff} result for the 3-of-4 criticality analysis reported in Table 7.1 of HI-2073758 are further broken down in the response to NRC Request 12.

Differences in Gadolinium content due to manufacturing tolerance are accounted for in the final maximum k_{eff} calculation in HI-2073758. That uncertainty, as documented in the response to NRC Request 12, is included herein as 0.00318 Δk_∞.

Critical experiment data are generally not available for spent fuel and, accordingly, some judgment must be used to assess those uncertainties introduced by the depletion calculations. CASMO-4 has been used extensively to generate assembly average cross sections for core follow calculations and reload fuel design in both BWRs and PWRs. Significant deviations between the predicted and actual fuel cycle lengths and core power distributions using CASMO-4 generated cross sections are not observed.

The burnup uncertainty due to the depletion of Gadolinium is not a large value and is difficult to quantify. Large conservatism in the assumption that all fuel lattices in the storage racks are at peak reactivity and that all of the Boraflex absorber panels are assumed to provide no reactivity hold down more than compensate for small unknown reactivity uncertainty values such as due to Gadolinium depletion.

Finally, assuming a burnup reactivity uncertainty, based upon five percent of the reactivity decrement between beginning of life (BOL) and peak reactivity for assembly depletion uncertainties (i.e., as discussed in Reference 3) of < 0.004 Δk specific to the bounding A10B-460L11G60 lattice in Table 6.1 of ANP-2684, the statistical combination of uncertainties presented in the response to NRC Request 12 would increase the statistically combined reactivity uncertainty from 0.0102 Δk to 0.0110 Δk. Even with this additional uncertainty, the maximum k_{eff} in Table 7.1 of HI-2073758 would remain the same because the reported uncertainty of 0.0111 Δk would not change.

NRC Request 14

Section 7.3, Effect of the Channel and Eccentric Fuel Positioning, of the Holtec criticality analysis discusses the uncertainties associated with the flow channel and eccentric positioning of the FA within a SFP storage cell.

a) The nominal calculations have the FA centered in the cell. For the eccentric positioning determination the analysis states, "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 discussion leaves out several pertinent details. Are they all pushed into the same corner? That would essentially keep the distance between them constant. Are they all pushed into the corner which brings them closest together? Since it is a 2x2 array modeled as infinite, even that essentially keeps the net separation the same. What if not all of the assemblies are pushed into the corner? Does it matter which ones are? What if the 3-of-4 storage configuration is surrounded by 4-of-4 storage configurations, or vice versa? Provide the necessary details for the staff to independently reach a reasonable assurance determination regarding eccentric fuel positioning.

b) The analysis concludes that centering the fuel assemblies in the cells is the most reactive configuration. Since the channel takes up space in the cell it effectively reduces the amount of eccentricity that can be achieved by the model. Would increased eccentricity affect the conclusion? Are these independent parameters?

Response

The eccentric positioning calculations consider that each group of three assemblies (i.e., 2x2) is clustered together as shown in Figure 3 below. Periodic boundary conditions simulate an infinite array of eccentrically located fuel assemblies, clustered in groups of three. The blocking device is not considered to be eccentrically positioned because movement of the blocking device within the cell would have a negligible effect on reactivity. Eccentric positioning does not result in an increase in reactivity because although the assemblies are placed closer to one another in one location, it creates a larger water gap between other adjacent assemblies. The distribution of 3-of-4 and 4-of-4 storage patterns in the SFP would not effect the conclusion that eccentric positioning does not increase the reactivity.

The presence of the channel would restrict the amount of eccentricity possible. However it should be noted that the absence of the channel would result in a decrease in the reactivity by 0.0076 Δk. Any reactivity effect of additional eccentricity because of the missing fuel channel is not expected to have any additional reactivity effect and would be offset by the absence of the channel.

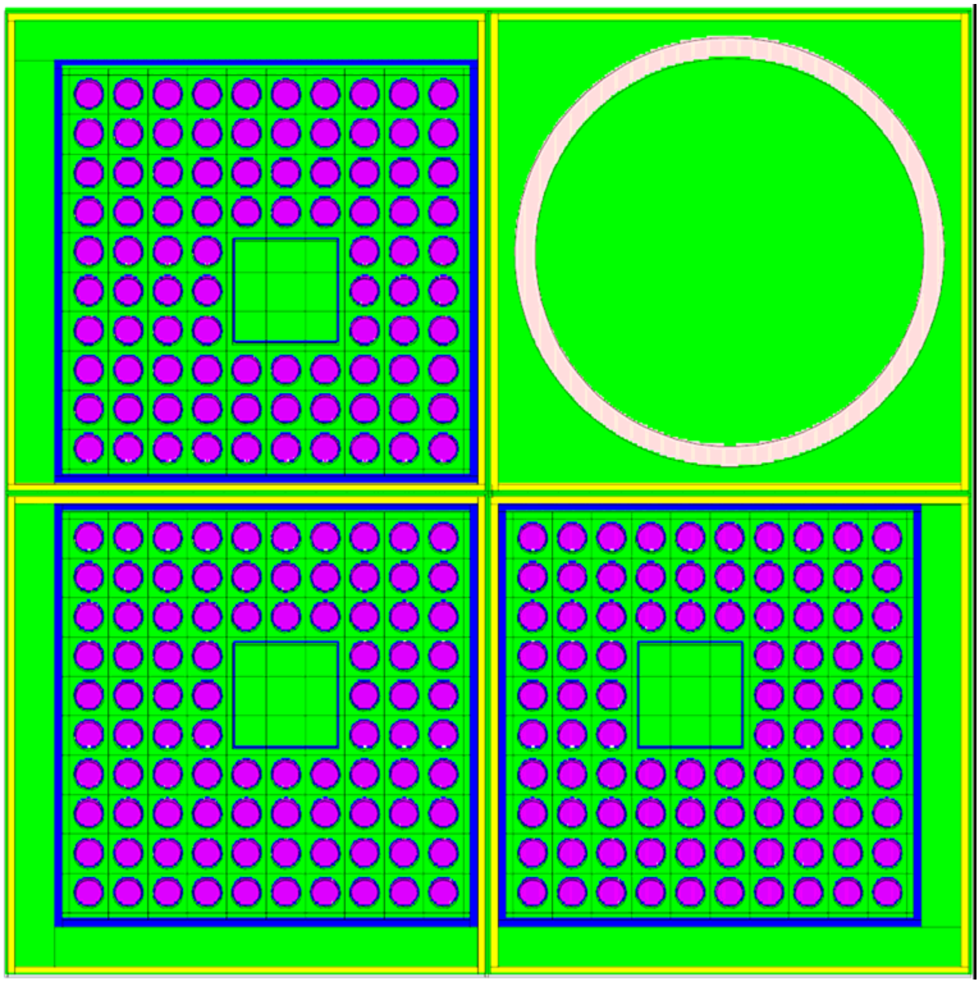

Figure 3: Calculational Model of Eccentric Positioning as Drawn by the Two-Dimensional Plotter in MCNP4a

NRC Request 15

Section 7.4, Effect of Fuel Assembly Orientation, of the Holtec criticality analysis discusses whether the orientation of the FA in the storage cells affects the reactivity. The ATRIUM-10 FA is asymmetric, in that the water hole is off center. The nominal analysis orients the water hole in the same place for all cells. This portion of the analysis indicates that sensitivity studies were performed to determine whether or not this had an effect. Provide the results from the sensitivity studies.

Response

The sensitivity studies for fuel assembly orientation were not included as part of the licensing report because: (1) there was no reactivity effect associated with fuel assembly orientation, (2) the results confirmed initial expectations, (3) the results are contained in the supporting calculational package, which is controlled by the Holtec QA program, and (4) an effort was made to not overly complicate the licensing report with analysis results that have very little impact on the overall conclusions.

Details are presented below showing the results of the sensitivity studies. Those studies show that the orientation of the assembly has a negligible effect on reactivity.

Effect of Assembly Rotation		
Case	**k_∞ In Rack**	**Δk**
Reference Case	0.9261	Reference
Pattern 1	0.9260	-0.0001
Pattern 2	0.9257	-0.0004
Pattern 3	0.9250	-0.0011
Pattern 4	0.9246	-0.0015
Pattern 5	0.9253	-0.0008
Pattern 6	0.9252	-0.0009
Pattern 7	0.9251	-0.0010
Pattern 8	0.9266	0.0005
Pattern 9	0.9249	-0.0012

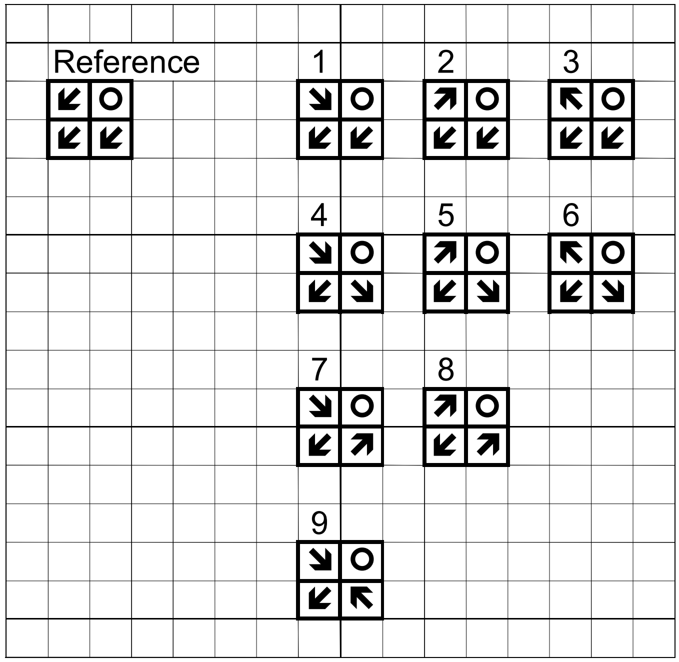

Figure 4: Effect of Assembly Rotation

NRC Request 16

Section 7.7, Abnormal and Accident Conditions, of the Holtec criticality analysis discusses which accidents are creditable and which are not.

a) According to the analysis, dropping one FA on top of another may result in small compression of the bottom FA. It comes with this statement, "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." Elsewhere they have determined that hotter water is more reactive. Hotter water reduces the water-to-fuel ratio. Why is a reduction in water-to-fuel ratio more reactive for the temperature increase but less reactive for the FA compression?

b) There is no reactivity discussion associated with this accident, just the potential damage to the rack or FA. Why would this accident have no reactivity impact?

Response

The description in the license amendment request (i.e., Reference 2) oversimplifies the reactivity effect of moderator temperature by attributing it to a change in the water-to-fuel ratio. There are other reactivity effects from a change in the moderator temperature that are not related to the water-to-fuel ratio, such as changes in the cross-sections (i.e., Doppler broadening). The calculations performed in response to NRC Request 10 show that the presence of zircaloy grid straps, which reduce the water-to-fuel ratio, results in a reduction in reactivity. Therefore, the reduction in water-to-fuel ratio due to fuel assembly compression would also result in a reduction in reactivity as stated in Section 7.7 of HI-2073758.

The discussion in HI-2073758 Section 7.7.1 of dropping a fuel assembly into an empty cell is in the context of dropping the fuel assembly into a storage cell that is empty, but is intended to contain a fuel assembly. The reactivity effect of this event is discussed in paragraph three of Section 7.7.1 of HI-2073758.

NRC Request 17

Section 7.8, Misloading a Fuel Assembly in a Location Intended to be Empty, of the Holtec criticality analysis discusses the reactivity affect of misloading a FA into a cell intended to have a blocking device. This section of the analysis shows that if a fuel assembly is loaded into a cell designated for a blocking device, then the regulatory requirement is exceeded.

a) Why is it creditable to drop a FA into an empty cell but not misload a FA into an empty cell?

b) As discussed previously, the staff considers misloading a fuel assembly in the SFP a creditable event. This analysis does not consider it a creditabe event. Explain why misloading a FA into a cell instead of a "blocking device" is not a creditable event.

Response

The reactivity effect of either dropping or misloading a fuel assembly into a storage cell intended to contain a blocking device is identical because the dropped assembly is assumed to land in the highest reactivity position as opposed to breaking through the bottom of the rack or not fully inserting into the cell. As indicated in the response to NRC Request 6, EGC will commit to using the blocking devices in a periodic repetitive pattern. Also as indicated in the response to NRC Request 6, prior to removing a blocking device from a cell, EGC will remove fuel to ensure that no more than two bundles remain in the affected 2x2 arrays. With this commitment for the locations of blocking devices being administratively controlled by procedure as discussed in response to NRC Request 1, it is not considered credible to misload a fuel assembly into a location intended for a blocking device. Having a blocking device in a location will prevent the dropping or misloading of a fuel assembly into that location. The response to NRC Request 1 provides detail regarding the diverse robust mechanical and administrative barriers to preventing the misloading of a fuel assembly into a cell designated for a blocking device. Table 7.5 of Attachment 3 of Reference 2 contains the results for a postulated mislocation and demonstrates 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. It should be further noted that this result is substantially conservative in that it reflects all fuel being modeled in the x-y-z directions at its most reactive condition; whereas, due to axial burnup profiles, such a reactivity condition will never occur in the SFP.

NRC Request 18

Section 1.0, Introduction, of the ANP criticality analysis (ANP-2684) states, "Reference 1 contains an evaluation of the spent fuel storage pool of the LSCS 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 percent checker-board loading) and no credit for Boraflex. The Reference 1 evaluation included the worst credible conditions and uncertainties." The analysis in ANP-2684 does not stand alone, rather it depends heavily on the work done in Reference 1, in several instances taking directly from the earlier work and applying them to the current work. As ANP-2684 develops the REFFE used in the Holtec analysis, ANP-2684 is essential to the review of the current LAR. Therefore, the information in ANP-2684 Reference 1 is critical for review of the current LAR. Provide ANP-2684 Reference 1.

Response

ANP-2684 cites EMF-2808(P) as Reference 1 to provide historical perspective in defining the bounding lattice (i.e., A10B-460L11G60) and because it is the predecessor to HI-2073758. The Holtec report addresses all necessary uncertainty and bias values, as supplemented by the response to NRC Request 12, and demonstrates that the 2.45 wt% U-235 ATRIUM-10 assembly model does not exceed the 0.95 k_{eff} regulatory limit. For this license amendment request, HI-2073758 serves as a substitute for EMF-2808(P); therefore, EMF-2808(P) does not need to be submitted.

NRC Request 19

The ANP criticality analysis indicates CASMO-4 was used for in-rack SFP storage rack analysis. ANP has a vendor specific topical report regarding the use of CASMO-4 for BWR analysis, 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, (Reference 6). This topical report is part of the LSCS list of approved methodologies. However, this topical report appears to be limiting to in-core analyzes.

a) Provide a LSCS site-specific justification for using CASMO-4 for SFP in-rack criticality analysis.

b) No code validation for the use of CASMO-4 was provided. It does not appear that CASMO-4 code bias and uncertainty were determined or applied. Provide these in accordance with References 2 and 3.

Response

The use of CASMO-4 in performing differential k_∞ calculations has been recognized by the NRC in the past and remains acceptable for the differential k_∞ calculations performed in support of both ANP-2684 and HI-2073758. Two recent reviews where the NRC recognized the use of CASMO in AREVA (ANP) criticality evaluations is shown in Section 3.2 of Enclosure 3 of Reference 6 and in Reference 7.

The use of CASMO-4 by AREVA for SFP licensing activities on BWR plants, and NRC approval of that use, provides the justification for using CASMO-4 for determination of in-rack assembly relative reactivity differences for the LSCS SFP 3-of-4 criticality analysis.

CASMO-4 has been benchmarked by Studsvik against critical experiments and MCNP as indicated in References 13 and 14 of EMF-2158(P)(A). These two References from EMF-2158(P)(A) are the same as References 3 and 4 of HI-2073758, and have previously been provided to the NRC as discussed in response to NRC Request 9. The average bias from the critical experiments is not large (i.e., <2 mk). Since CASMO-4 has only been used by AREVA in ANP-2684 for either sensitivity analyses or to calculate comparative k_∞ values, the code bias does not come into play and is therefore not listed in ANP-2684.

<u>NRC Request 20</u>

The ANP criticality analysis indicates there is no significant difference with the FA channeled or not. This is opposite to the conclusion drawn in the Holtec analysis for the same fuel assembly in the same configuration. Explain this difference.

<u>Response</u>

The phrase "fuel channel configurations" used at the end of Section 4.2 of ANP-2684 is intended to address the different types of fuel channels that might be placed on a fuel assembly, (e.g., 80 mil, 100 mil, or advanced channel with "thin" walls and "thick" corners). AREVA concurs with Holtec's statements regarding channeled and unchanneled assemblies (i.e., including the channel in the analysis is conservative).

<u>NRC Request 21</u>

The ANP criticality analysis makes no attempt to establish uncertainties as specified in references 2 and 3. As the ANP criticality analysis is determining the REFFE those uncertainties are important and can and will change with different fuel assembly parameters. Provide the uncertainties and their basis. Include sufficient detail for the staff to independently reach a reasonable assurance determination regarding each uncertainty.

<u>Response</u>

ANP-2684 provides a relative reactivity comparison analysis that justifies the use of a single lattice assembly model at 2.45 wt% U-235 as bounding in an in-rack configuration and was not intended to independently support the licensing amendment with the establishment of an absolute k_{eff} value. Therefore, that document does not report biases and uncertainties. Report HI-2073758 calculates the maximum k_{eff} in Table 7.1, including uncertainties as well as code and modeling bias values, which follows NRC guidance in Reference 3. The response to NRC Request 12 provides additional detail relative to the uncertainty parameters considered and the reactivity impact of that uncertainty parameter.

NRC Request 22

The ANP criticality analysis makes no attempt to establish biases as specified in references 2 and 3. As the ANP criticality analysis is determining the REFFE, those biases are important and can and will change with different fuel assembly parameters. Provide the uncertainties and their basis. Include sufficient detail for the staff to independently reach a reasonable assurance determination regarding each uncertainty.

Response

Biases and uncertainties are discussed above in response to NRC Request 21.

NRC Request 23

The ANP criticality analysis provides no evidence to support the conclusions in Items 5 and 6 of Table 2.1. Provide the evidence to support these conclusions. Include sufficient detail for the staff to independently reach a reasonable assurance determination.

Response

In Table 2.1 of ANP-2684, items 5 and 6 are supported by the k_∞ comparisons in Tables 6.1 and 6.2. Item 5 refers to the maximum CASMO-4 k_∞ value of the A10B-460L11G60 lattice (i.e., 1.0981). This "bounding lattice" provides an acceptable in-rack CASMO-4 k_∞ limit because it is less reactive than the REBOL A10B-245L-0G0 lattice that is used to model a simple assembly in the HI-2073758 report. Item 6 refers to the maximum CASMO-4 k_∞ value of the as fabricated A10T-4444L-12G40 lattice (i.e., 1.1230), which is in use at LSCS. Use of this more reactive lattice for 12" or less below the top natural blanket is supported by the KENO sensitivity calculation presented in Table 6.2 (see the response to NRC Request 25). Adherence to the requirements of items 5 and 6 ensures that Case 2 of Table 6.2 of ANF-2684 remains bounding, thereby validating the continual use of the simple 2.45 wt% assembly model used in the Holtec HI-2073758 report.

NRC Request 24

The LAR does not describe how the limitations inherent in conclusions in Items 5 and 6 of Table 2.1 of the ANP criticality analysis are captured and controlled. These restrictions appear to be similar in function to burnup/enrichment loading curves that are included in PWR TSs. Explain how the proposed LSCS TS ensures these limitations are not exceeded.

Response

Section 4.3.1.1.a of the Technical Specifications provides the criteria for criticality compliance that k_{eff} be less than or equal to 0.95 for the SFP. Section 9.1.2.2.3 of the UFSAR contains the requirements for criticality compliance. Upon approval of the proposed license amendment request, the UFSAR will be updated as part of EGC's implementation process to include the interface requirements and criticality analysis documentation associated with the 3-of-4 criticality licensing basis, including the specific requirements for geometric arrangements, limitations on lattice enrichments, limitations on gadolinia content, etc.

NRC Request 25

Table 6.1 of the ANP criticality analysis indicates the maximum in-rack reactivity of lattice A10T-4444L-12G40 exceeds that of REBOL A10B-245L-0G0 and A10B-245L-0G0. How can the ATRUIM-10 2.45 uniform planar enrichment with no gadolinium be limiting?

Response

Table 6.1 of ANP-2684 establishes that the k_∞ of a REBOL A10T-270L-0G0 lattice (i.e., Case 14) exceeds that of the A10T-4444L-12G40 lattice (i.e., Case 10). Both of these results are based on the true configuration in the zone below the top natural U-blanket (i.e., reflects the presence of 83 fuel rods). Table 6.2 defines two KENO V.a calculations. Case 1 uses a single 2.45 wt% U-235 lattice with 91 fuel rods for the entire length of the assembly. Case 2 (i.e., the realistic case) models 83 or 91 fuel rods, as applicable to the ATRIUM-10 geometry, and uses 2.45 wt% lattices with top and bottom natural blankets and a 12" zone at 2.70 wt% U-235 just below the top natural blanket. The results in Table 6.2 demonstrate that a realistic fuel assembly with a high reactivity 12" fuel zone (e.g., the A10T-4444L-12G40 lattice) will have a lower k_∞ than the simple 2.45 wt% model as defined by Case 1 in Table 6.2. The comparison in Table 6.2 shows that the simple 2.45 wt% model (i.e., Case 1) bounds all assemblies using the lattices in Table 6.1.

NRC Request 26

Describe the process used to determine that fuel assemblies have attained proper BU for storage in the BU dependent racks.

Response

The 3-of-4 criticality analysis documented in HI-2073758 uses an ATRIUM-10 design that, in an in-rack configuration without Boraflex, is more reactive than, or justified as more reactive than, all fuel designs over all burnup ranges of all product lines used at LSCS. Therefore, there is no burnup dependency for storage of fuel in a 3-of-4 configuration at LSCS, and there are no fuel burnup dependent racks in the LSCS spent fuel storage pool.

NRC Request 27

Describe the process used to control movement of items within the SFP.

Response

As discussed in the response to NRC Request 1, the cells identified as degraded and the cells requiring blocking devices would be documented in a DCP under the EGC configuration control process. The DCP would include document changes to affected plant drawings, procedures, and databases reflecting the selected locations of the blocking devices and degraded cells.

EGC's Special Nuclear Material (SNM) Control and Accountability process governs the control and accounting of SNM. Move sheets control the movement of non-fuel components, SNM instruments and SNM sources, as well as fuel, where the beginning and ending locations are located within the site, including within the SFP. Move sheets document specific directions for movement of SNM items and non-fuel components, including initial location and location to which the item is to be moved. The move sheets are prepared and independently reviewed by a qualified reactor engineer and are used by the SNM Handlers during execution. The movements must comply with SFP constraints and other physical constraints, such as unusable SFP locations, including depleted cells due to Boraflex degradation, that are delineated in the SNM Control and Accountability process. Handling constraints are also in place for non-fuel components, such as blocking devices. The blocking devices that will be used as robust barriers will have a lifting eye for handling with special tooling, such as a J-hook. The J-hook is not intended for everyday use or for the movement of fuel assemblies. Therefore, inadvertent removal of the blocking device with the refuel bridge is precluded since the blocking device will not have a bail handle and can only be moved with a J-hook. In addition, administrative processes and procedures govern the use of a J-hook and provide a second barrier to inadvertent removal of a blocking device. Move sheet packages include restrictions and special instructions for the SNM Handlers and dedicated emergency set-down locations. Move sheets involving the movement of fuel must be generated using approved and controlled move planning software (i.e., SHUFFLEWORKS). The move planning software is controlled using the Digital Technology Software Quality Assurance process. Appropriate constraints must be set in the software before generating move sheets, which will include locations of degraded cells and blocking devices. These locations will be appropriately updated with movement constraints to prevent inadvertent misloading and movement within the SFP. The move planning software database is updated after the execution of the move sheets to reflect the movement of items and their final locations. During the movement of SNM and non-fuel components, a fuel movement spotter provides independent verification of each move. This consists of independent verification of proper bridge location for both picking up and storing of items in the designated location on the move sheet.

References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

2. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "License Amendment Request Regarding Spent Fuel Storage Pool Criticality," dated December 13, 2007

3. Memorandum from L. Kopp (U. S. NRC) to T. Collins (U. S. NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 19, 1998

4. Letter from J. A. Umbarger (Studsvik Scandpower, Inc.) to U. S. NRC, "Transmittal of Copies of CASMO-4 Benchmark Reports Relevant to EMF-2158(P) Revision 0 'Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2,'" dated April 30, 1999

5. NUREG/CR-6683, "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"

6. Letter from S. N. Bailey (U. S. NRC) to J. Scarola (Carolina Power & Light Company), "Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendment Storage of AREVA NP Fuel (TAC Nos. MD4061 and MD4062)," dated November 27, 2007

7. Letter from C. P. Patel (U. S. NRC) to C. J. Gannon (Carolina Power & Light Company), "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Regarding Soluble Boron Credit for Fuel Storage Pools (TAC No. MC8267)," dated March 10, 2006

ATTACHMENT 2
Regulatory Commitments

The following list identifies those actions committed to by Exelon Generation Company, LLC, (EGC) for LaSalle County Station in this submittal. Any other actions discussed in the submittal represent intended or planned actions by EGC, are described only for information, and are not regulatory commitments.

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (YES/NO)	PROGRAM-MATIC (YES/NO)
To simplify implementation of the placement of blocking devices for the 3-of-4 criticality analysis, EGC will use the blocking device in a periodic repetitive pattern such that all blocking devices would be installed in every other cell in every other row of the spent fuel storage pool sufficient to cover the depleted cells in a 3-of-4 configuration. The coverage would be extended sufficiently in the x and y directions to cover the depleted cells of interest such that the 3-of-4 configuration requirements are met.	Upon implementation of license amendment	No	Yes
EGC will implement procedural controls to require that prior to removing a blocking device from a cell, fuel will be removed to ensure that no more than two bundles remain in the affected 2x2 arrays. This will ensure that a single misloaded bundle will not create a condition not already analyzed by the 3-of-4 analysis.	Upon implementation of license amendment	No	Yes

ATTACHMENT 3
Revised 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 PAGE

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.

c. For Unit 2 only, for any 2x2 arrangement of storage cells for which the Boraflex neutron absorber cannot maintain the requirements of 4.3.1.1.a, a blocking device shall be installed in one of the four cells of the affected 2x2 array.

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