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Waterford 3

Attachment 3 contains Proprietary Information

W3F1-2008-0052

September 17, 2008

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

SUBJECT: License Amendment Request NPF-38-277
License Amendment Request to Modify
Technical Specification Section 5.6, Fuel Storage and Add New Technical
Specification 3/4 9.12, Spent Fuel Pool Boron Concentration
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment to Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications (TS). The Waterford 3 TS are being revised to take credit for soluble boron in Region 1 (cask storage pit) and Region 2 (spent fuel pool and refueling canal) fuel storage racks for the storage of both Standard and Next Generation Fuel (NGF) assemblies. In accordance with 10CFR 50.68, the limits for k_{eff} of the spent fuel storage racks are appropriately revised based on analysis to maintain k_{eff} less than 1.0 when flooded with unborated water, and less than, or equal to, 0.95 when flooded with water having a minimum boron concentration of 447 ppm, during normal conditions. A new Technical Specification is added which includes a surveillance that ensures the required boron concentration is maintained in the spent fuel storage racks. This added Technical Specification Surveillance conforms to the guidance of NUREG-1432. The change is evaluated for both normal operation and accident conditions. This change will provide more flexibility in storing the more reactive NGF assemblies in the spent fuel storage racks.

This proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that this change involves no significant hazards consideration. The bases for these determinations are included in the attached submittal.

There are no new regulatory commitments contained in this submittal.

Entergy requests NRC approval of the proposed amendment by September 10, 2009, in order to support the Fall 2009 planned refueling outage. Once approved, the amendment will be implemented within 60 days of receipt.

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Attachment 1 contains an analysis of the proposed TS change. Attachment 2 contains the proposed TS changes. Attachment 3 documents the criticality safety evaluation for the storage of Standard and Next Generation Fuel assemblies in Holtec-designed Region 1 and Region 2 style high-density spent fuel storage racks (SFSRs) at Waterford Unit 3, and the evaluation provides the analytical basis for the Technical Specification changes. Attachment 3 contains proprietary information. The proprietary information was provided to Entergy in a Holtec International transmittal that is referenced by an affidavit. Holtec requests the enclosed proprietary information identified in Attachment 3 be withheld from public disclosure in accordance with the provisions of 10CFR 2.390 and 10CFR 9.17. Attachment 4 contains the affidavit for withholding the proprietary information contained in Attachment 3. Attachment 5 contains the Non-Proprietary Licensing Report documenting the criticality safety evaluation for the storage of Standard and Next Generation Fuel assemblies in Holtec-designed Region 1 and Region 2 style high-density spent fuel storage racks (SFSRs) at Waterford Unit 3.

Please contact Robert J. Murillo, Manager, Licensing at (504) 739-6715 if there are any questions concerning this matter.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 17, 2008.

Sincerely,

G. Kenlet for K. Walsh

KTW/GCS/ssf

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3 Holtec Report: Licensing Report for Waterford 3 NGF Criticality Analysis (Proprietary Information)
4 Affidavit for withholding Proprietary Information
5. Holtec Report: Licensing Report for Waterford 3 NGF Criticality Analysis (Non Proprietary Information)

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Attachment 1

to

W3F1-2008-0052

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3). The Waterford 3 TS are being revised to take credit for soluble boron in the Region 1 (cask storage pit) and Region 2 (spent fuel pool and refueling canal) fuel storage racks for the storage of both Standard and Next Generation Fuel (NGF) assemblies. In accordance with 10 CFR50.68, the limits for k_{eff} of the spent fuel storage racks are appropriately revised based on analysis to maintain k_{eff} less than 1.00 when flooded with unborated water, and less than, or equal to, 0.95 when flooded with water having a minimum boron concentration of 447 ppm during normal conditions. A new Technical Specification is added which includes a surveillance that ensures the required boron concentration is maintained in the spent fuel storage racks. The boron concentration will be maintained at a minimum of 1900 ppm per the new Technical Specification, significantly exceeding the required concentration levels to maintain k_{eff} within regulatory requirements. The change is evaluated for both normal operation and accident conditions. This change will provide more flexibility in storing the more reactive NGF assemblies in the spent fuel storage racks. This change does not involve a Significant Hazards Consideration, and the change is in conformance with regulatory requirements.

2.0 PROPOSED CHANGE

The proposed change will modify TS section 5.6 as follows:

a. TS 5.6.1a wording will be changed to: "For Region 1 (cask storage pit) and Region 2 (spent fuel pool and refueling canal) racks, a maximum k_{eff} of less than 1.00 when flooded with unborated water, and less than, or equal to, 0.95 when flooded with water having a boron concentration of 447 ppm."

b. TS 5.6.1d will be revised to replace the words "New or partially spent" with "Fresh and irradiated" and will read as follows: "Fresh and irradiated fuel assemblies may be allowed unrestricted storage in Region 1 racks."

c. TS 5. 6.1e will be revised to replace the word "New" with "Fresh" and will read as follows: "Fresh fuel assemblies may be stored in the Region 2 racks provided that they are stored in a "checkerboard pattern" as illustrated in Figure 5.6-1."

d. TS 5.6.1f will be revised to replace the words "Partially spent" with "Irradiated" and delete the word "discharge" and will read as follows: "Irradiated fuel assemblies with a burnup in the "acceptable range" of Figure 5.6-2 may be allowed unrestricted storage in the Region 2 racks."

e. TS 5.6.1g will be revised to replace the words "Partially spent" with "Irradiated" and delete the word "discharge" and replace the word "spent" with "irradiated" and will read as follows: "Irradiated fuel assemblies with a burnup in the "unacceptable range" of Figure 5.6-2 may be stored in the Region 2 racks provided that they are stored in a "checkerboard pattern," as illustrated in Figure 5.6-1 with fuel in the "acceptable range" of Figure 5.6-3."

f. TS 5.6.2 will be revised to replace the word "new" with "fresh" and will read as follows: "The k_{eff} for fresh fuel stored in the new fuel storage racks shall be less than or equal to 0.95 when flooded with unborated water and shall not exceed 0.98 when aqueous foam moderation is assumed."

g. TS Figure 5.6-1 will be replaced by a new Figure 5.6-1 to show the new Alternative Checkerboard Storage Arrangements.

h. TS Figure 5.6-2 will be replaced by a new Figure 5.6-2 to show the new Acceptable Burnup Domain for Unrestricted Storage of Irradiated Fuel in Region 2 of the Spent Fuel Pool.

i. TS Figure 5.6-3 will be replaced by a new Figure 5.6-3 to show the Acceptable Burnup Domain for Irradiated Fuel in a Checkerboard Arrangement with Fuel of 5 wt % Enrichment, or less, at 27 GWd/MTU Burnup, or higher, in Region 2 of the Spent Fuel Pool.

j. A new TS Figure, TS Figure 5.6-4, will be added to show Examples of Contiguous Fresh and Irradiated Fuel Checkerboards Which Meet Interface Requirements.

The proposed change will add TS 3/4.9.12 as follows:

1. TS 3/4.9.12 will be entitled "Spent Fuel Pool (SFP) Boron Concentration."

2. A Limiting Condition for Operation will read as follows: "3.9.12 The spent fuel pool boron concentration shall be $\geq$ 1900 ppm."

3. The Applicability will read as follows: "When fuel assemblies are stored in the SFP."

4. The Action statement will read as follows: "With the spent fuel boron concentration not within limits immediately suspend movement of fuel in the SFP and immediately initiate actions to restore boron concentration to within limits."

5. The Surveillance Requirements will read as follows: "4.9.12 Verify the spent fuel pool concentration is within limits once per 7 days."

3.0 BACKGROUND

License Amendment 214, approved on May 9, 2008, allowed the use of Next Generation Fuel (NGF) for Waterford 3. The new NGF fuel assemblies have a higher fuel pellet density, smaller rod diameter and thinner fuel rod cladding which results in the NGF fuel assembly being more reactive than the current Standard fuel assemblies. The acceptable storage patterns of the NGF assemblies in the spent fuel storage racks are currently limited due to the higher reactivity of these assemblies. The proposed TS changes will provide more flexibility in the storage pattern for NGF stored in the spent fuel storage racks by taking credit for soluble boron to ensure that k_{eff} remains within regulatory limits. Criticality analysis has demonstrated that taking credit for soluble boron in the spent fuel storage racks will ensure that k_{eff} remains within regulatory limits.

The purpose of the spent fuel storage racks is to maintain the fresh and irradiated assemblies in a safe storage condition. The current licensing basis as defined by the existing Technical Specification Requirements, Section 5.6, and federal code requirements, 10 CFR 50.68, specify the normal and accident parameters associated with maintaining the fresh and irradiated assemblies in a safe storage condition. Per the existing Technical Specification, the k_{eff} of the spent fuel storage racks are designed to be maintained less than or equal to 0.95 when flooded with unborated water. 10 CFR 50.68 defines the criticality accident requirements associated with the spent fuel racks and states the following: "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% probability, 95% confidence level, if flooded with unborated water. If credit is taken for soluble boron, 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% probability, 95% confidence level, if flooded with borated water, and the k_{eff} must remain below 1.0, subcritical, at a 95% probability, 95% confidence level, if flooded with unborated water."

Waterford 3's current Technical Specification does not take credit for soluble boron to maintain $k_{eff} \leq 0.95$. Accordingly, Waterford 3 is in compliance with 10 CFR 50.68 which states "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% probability, 95% confidence level, if flooded with unborated water." The analysis shows that a minimum soluble boron concentration of 447 ppm is required to maintain k_{eff} within the regulatory requirement of ≤ 0.95. Based on the proposed amendment, which will credit boron to maintain regulatory conformance, the following excerpt from 10 CFR50.68 applies: "If credit is taken for soluble boron, 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% probability, 95% confidence level, if flooded with borated water, and the k_{eff} must remain below 1.00, subcritical, at a 95% probability, 95% confidence level, if flooded with unborated water." The proposed applicable Technical Specification change is in compliance with the above statement and reads as follows: "For Region 1 and Region 2 racks, a maximum k_{eff} of less than 1.0 when flooded with unborated water, and less than, or equal to 0.95 when flooded with water having a boron concentration of 447 ppm."

An analysis, provided in Attachment 3 as a Holtec International Report, demonstrated that the effective neutron multiplication factor, k_{eff}, is less than 1.00 with the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with unborated water at a temperature corresponding to the highest reactivity. In addition, the analysis demonstrated that k_{eff} is less than or equal to 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with borated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity included a margin for uncertainty in reactivity calculations including manufacturing tolerances and is shown to be less than 0.95 with a 95% probability at a 95% confidence level. Reactivity effects of abnormal and accident conditions were also evaluated to assure that under all credible abnormal and accident conditions, the k_{eff} will not exceed the regulatory limit of 0.95 under borated conditions or 1.0 with unborated water.

4.0 TECHNICAL ANALYSIS

Holtec Report No. HI-2084014, entitled "Licensing Report for Waterford Unit 3 Spent Fuel Pool Criticality Analysis" (Attachment 3) provides the technical analysis for the proposed change to store Standard and Next Generation Fuel (NGF) assemblies in Holtec-designed Region 1 and Region 2 style high-density spent fuel storage racks at Waterford 3. The report analyzed the impact of the change on Region 1 and Region 2 of the spent fuel storage racks and the resultant interfaces within and between the racks in each region. Also, the report performed and evaluated various calculations related to the Fuel Transfer Carriage Criticality, Upender Criticality, New Fuel Elevator Criticality, Boron Dilution Accident Evaluation, Low Flow Rate Dilution, High Flow Dilution, Temporary Storage Racks (in the refueling pool inside containment), Fuel Pin Storage Container, and New Fuel Storage Vault.

The results of the analysis determined that the high-density spent fuel storage racks for Waterford 3 were designed using applicable codes and standards. The analysis showed that the effective neutron multiplication factor, k_{eff}, is less than 1.00 with the racks fully loaded with the fuel of the highest anticipated reactivity, and flooded with unborated water at a temperature corresponding to the highest reactivity. The report demonstrated that k_{eff} is less than or equal to 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with borated water at a temperature corresponding to the highest reactivity. Also, reactivity effects of abnormal and accident conditions will not result in k_{eff} exceeding the regulatory limit of 0.95 under borated conditions.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

By letter dated February 9, 1983, the Nuclear Regulatory Commission issued a Material License (no. SNM-1913) to Waterford 3 which authorizes the receipt, possession, inspection, and storage of uranium enriched in the U-235 isotope, contained in fuel assemblies, and the receipt, possession, and use of two Pu-Be neutron sources. In the letter, the NRC granted Waterford 3 an exemption (related to criticality alarm systems) from the requirements of 10 CFR 70.24, *"Criticality Accident Requirements."* With the approval of the proposed change this exemption is no longer required. Waterford 3 currently complies with the requirements of 10 CFR 50.68, *"Criticality Accident Requirements."*

Waterford 3 proposed change will comply with the requirements of 10 CFR 50.68, *"Criticality Accident Requirements."*

There are eight criteria that must be satisfied in the regulation. Waterford 3 complies with these as follows:

1) 10 CFR 50.68, (b) (1) - Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

W3's fuel handling procedures ensure that subcriticality is maintained in the reactor vessel and the spent fuel storage racks under the most adverse moderation conditions feasible by unborated water. Storage of fuel assemblies is procedurally controlled to assure k_{eff} remains below 1.0, at a 95% probability, 95% confidence level, when flooded with unborated water.

2) 10 CFR 50.68, (b) (2) - The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

   Criticality calculations of the new fuel vault fully loaded with Standard and NGF fresh fuel assemblies and filled with the most reactive unborated water showed that reactivity did not exceed 0.95, at a 95% probability, 95% confidence level.

3) 10 CFR 50.68, (b) (3) - If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

   Criticality calculations were performed on the new fuel vault fully loaded with Standard and NGF fresh fuel assemblies and filled with the most reactive low density hydrogenous fluid. The results of these calculations showed that reactivity did not exceed 0.98, at a 95% probability, 95% confidence level.

4) 10 CFR 50.68, (b) (4) - If no credit for soluble boron is taken, the k-effective 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. If credit is taken for soluble boron, 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 borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

   Soluble boron credit will be taken in the spent fuel storage racks. Reactivity will not exceed 0.95 at a 95% probability with a 95% confidence level with at least 447 ppm boron. The criticality calculations included in the proposed change show that k_{eff} remains below 1.0, at a 95% probability, 95% confidence level, when flooded with unborated water.

5) 10 CFR 50.68, (b) (5) - The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

W3 does not currently have a quantity of SNM, other than the nuclear fuel, stored on site to establish a critical mass.

6) 10 CFR 50.68, (b) (6) - Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

   Radiation monitors are located in the spent fuel storage area which alarm in the control room. When fuel movement is in progress additional radiation monitors are placed directly on the fuel handling bridges to provide an additional audible indication of excessive radiation levels.

7) 10 CFR 50.68, (b) (7) - The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

   Per W3 TS 5.6.1 h, the maximum U-235 fuel enrichment limit is 5.0 weight percent.

8) 10 CFR 50.68, (b) (8) - The FSAR is amended no later than the next update which 50.71(e) of this part requires, indicating that the licensee has chosen to comply with 50.68(b).

   The W3 FSAR will be amended no later than the next required update after the proposed TS change is approved and implemented.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any General Design Criterion (GDC) differently than described in the Updated Final Safety Analysis Report (UFSAR).

5.2 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 to take credit for soluble boron and revise the loading patterns in the Region 1 (cask storage pit) and Region 2 (main storage pool) of the spent fuel storage racks for the storage of Standard and Next Generation Fuel (NGF) assemblies will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The following potential accident scenarios have been evaluated:

   - Dropped assembly – horizontal
   - Dropped assembly – vertical
   - Misloaded fresh assembly
   - Mislocated fresh fuel assembly

The proposed change in criticality limits will not increase the probability of any of these potential accidents.

For the situation in which a fuel assembly is assumed to be dropped above a spent fuel storage rack and come to rest horizontally on top of the rack, the minimum separation distance between the dropped assembly and the top of the active fuel region of assemblies in the racks would be more than 12 inches, which would be neutronically an infinite separation, thereby precluding a significant increase in reactivity.

A vertical drop of an assembly onto another assembly in a spent fuel storage rack has been shown to cause no significant damage to either fuel assembly. A vertical drop into an empty storage cell could result in a small, localized deformation in the rack baseplate which could produce a misalignment between the active fuel region of the dropped assembly and the neutron absorbing Boral of the rack. The corresponding reactivity increase would be small, and would be bounded by the reactivity increase resulting from the misloading of a fresh fuel assembly in the Region 2 racks.

The Region 1 racks have been shown analytically to be qualified for the storage of fresh fuel assemblies with a maximum enrichment of 5.0 wt% U-235. Therefore, the misloading of a fuel assembly within the Region 1 racks is not a concern.

The inadvertent misloading of a fresh fuel assembly into a Region 2 storage cell which was intended for the storage of an irradiated fuel assembly would not result in exceeding the regulatory k_{eff} limit of 0.95 if a soluble boron level of 838 ppm or more were present. The concentration of boric acid in the water during fuel movement is maintained $\geq$ 1900 ppm in accordance with Technical Specification 3/4.9.12.

The mislocation of a fresh fuel assembly with a maximum enrichment of 5.0 wt% U-235 outside of a Region 1 or Region 2 rack and adjacent to other fuel assemblies would not result in exceeding the regulatory k_{eff} limit of 0.95 if a soluble boron level of 534 ppm or more were present.

Therefore, it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

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

Response: No

The proposed change to take credit for soluble boron and revise the loading restrictions in the Region 1 (cask storage pit) and Region 2 (main storage pool) of the spent fuel storage racks for the storage of Standard and Next Generation Fuel (NGF) assemblies will not create the possibility of a new or different kind accident from any accident previously evaluated.

Soluble boron has been maintained in the Region 1 and Region 2 water and is currently required by procedures. Therefore, crediting soluble boron in the spent fuel storage rack criticality analysis will have no effect on normal pool operation and maintenance. Crediting soluble boron will only result in increased sampling to verify the boron concentration in accordance with new TS 3/4/9.12. This increased sampling ensures that a new kind of accident not previously evaluated, boron dilution in the spent, is not created. A dilution of the spent fuel storage rack soluble boron has always been a possibility. However, the boron dilution event previously had no consequences, since boron was not previously credited in the accident analysis. The initiating events that were considered for having the potential to cause dilution of the boron in the spent fuel storage pool to a level below that credited in the criticality analyses fall into three categories: dilution by flooding, dilution by loss-of-coolant induced makeup, and dilution by loss-of-cooling system induced makeup. The addition of large amounts of unborated water would be necessary to reduce the boron concentration from the normal level of $\geq$1900 ppm to either 838 ppm or 447 ppm. This amount of water would be detected by the Operator and secured prior to reaching these boron concentrations.

A small dilution flow might result from a leak from the cooling system into the spent fuel pool. A dilution flow of 2 gpm from in-leakage might not be immediately detected, but would require more than 135 days to reduce the boron concentration in the spent fuel storage racks to the minimum required 447 ppm under normal conditions, and more than 72 days to reach the 838 ppm which would be required to accommodate the most limiting fuel misloading accident. These time periods are based on a conservative starting point of 1720 ppm boron. It is expected that routine surveillance measurements of the soluble boron concentration conducted every 7 days per new TS 3/4.9.12 would readily detect the reduction in concentration and provide sufficient time for corrective action prior to exceeding the regulatory limits.

The continuous operation of the Condensate Storage Pool makeup pump could add a large amount of unborated water to the spent fuel pool. Conservatively assuming instantaneous mixing of unborated water with the pool water, it would take approximately 648 minutes to reduce the soluble boron concentration to 447 ppm which is the minimum concentration required to maintain k_{eff} below 0.95 under normal operating conditions. During this dilution accident, 389,000 gallons of water would be released into the spent fuel storage racks. For this high flow rate scenario, 346 minutes would elapse before reaching the 838 ppm concentration which is the level needed to address the most limiting fuel misloading accident. These time periods are also based on a conservative starting point of 1720 ppm boron.

A high flow rate dilution accident would result in multiple alarms alerting the control room to the situation, including the fuel pool high-level alarm, Fuel Handling Building sump high level alarm, and the Liquid Waste Management Trouble alarm. It is not considered to be credible that multiple alarms would fail or be ignored by Operators in the control room. Spilling of large volumes of water from the spent fuel storage pool would be observed by plant personnel during these calculated time periods and result in corrective actions prior to exceeding regulatory limits.

In the unlikely event that soluble boron in the spent fuel storage racks is completely diluted, the fuel in the spent fuel storage racks will remain subcritical by a design margin of at least 0.005 Δk, and so the k_{eff} of the fuel in the spent fuel storage racks will remain below 1.00.

The proposed change will not result in any other change in the plant configuration or equipment design.

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

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

Response: No

The proposed change to take credit for soluble boron and revise the loading restrictions in the Region 1 (cask storage pit) and Region 2 (main storage pool) of the spent fuel storage racks for the storage of Standard and NGF assemblies does not involve a significant reduction in a margin of safety.

Detailed analysis with approved and benchmarked methods has shown with a 95% probability at a 95% confidence level that the neutron multiplication factor, k_{eff}, of the Region 1 and Region 2 high-density spent fuel storage racks loaded with either Standard or NGF assemblies, and including biases, tolerances, and uncertainties, is less than 1.00 with unborated water, and less than 0.95 with 447 ppm of soluble boron credited. In addition, the effects of abnormal and accident conditions have been evaluated to demonstrate that under credible conditions the k_{eff} will not exceed 0.95 with soluble boron credited. To ensure that the margin of safety for subcriticality is maintained and that k_{eff} will be below 0.95, a new TS requires a soluble boron level of $\geq$ 1900 ppm in the spent fuel pool. This is much greater than the required soluble concentration of 447 ppm under normal conditions, and 838 ppm for all credible accident conditions.

Therefore, it is concluded that the proposed changes do not involve a significant reduction in the margin of safety.

5.3 Environmental Considerations

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(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The NRC has approved similar submittals for ANO-2 and Calvert Cliffs in NRC SERs dated 9/30/2003 and 6/3/2004, respectively.

7.0 REFERENCES

a. Entergy License Amendment Request to Support Next Generation Fuel, Letter W3F1-2007-0037, dated August 2, 2007.
b. NRC letter dated May 9, 2008, Correction to Amendment No. 214 Re: Request to Support Next Generation Fuel, Review and Approval of Revised Emergency Core Cooling System (ECCS) Performance Analysis, and Review and Approval of Supplement to the ECCS Performance Analysis.
c. NRC letter to Louisiana Power and Light Company dated February 9, 1983 granting Waterford 3 exemption to 10 CFR 70.24 requirements.

Attachment 2

W3F1-2008-0052

Proposed Technical Specification Changes (mark-up)

DESIGN FEATURES

5.6 FUEL STORAGE

CRITICALITY

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

a. A normal k_{eff} of less that or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties. Replace with "A"

b. A nominal 10.185 inch center-to-center distance between fuel assemblies placed in Region 1 (cask storage pit) spent fuel storage racks.

c. A nominal 8.692 inch center-to-center distance between fuel assemblies in the Region 2 (spent fuel pool and refuelling canal) racks, except for the four southern-most racks in the spent fuel pool which have an increased N-S center-to-center nominal distance of 8.892 inches.

d. Fresh and irradiated ~~New or partially spent~~ fuel assemblies may be allowed unrestricted storage in Region 1 racks.

e. Fresh ~~New~~ fuel assemblies may be stored in the Region 2 racks provided that they are stored in a "checkerboard pattern" as illustrated in Figure 5.6-1.

f. Irradiated ~~Partially spent~~ fuel assemblies with a ~~discharge~~ burnup in the "accceptable range" of Figure 5.6-2 may be allowed unrestricted storage in the Region 2 racks.

g. Irradiated ~~Partially spent~~ fuel assemblies with a ~~discharge~~ burnup in the "unacceptable range" of Figure 5.6-2 may be stored in theRegion 2 racks provided that they are stored in a "checkerboard pattern", as illustrated in Figure 5.6-1, with ~~spent~~ irradiated fuel in the "acceptable range" of Figure 5.6-3.

h. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.

5.6.2 The k_{eff} for ~~new~~ fresh fuel stored in the new fuel storage racks shall be less than or equal to 0.95 when flooded with unborated water and shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation +40.0 MSL. When fuel is being stored in the cask storage pit and/or the refueling canal, these areas will also be maintained at +40.0 MSL.

CAPACITY

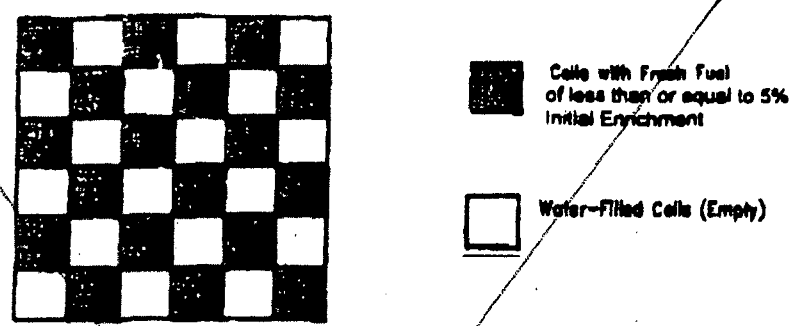
5.6.4 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1849 fuel assemblies in the main pool, 255 fuel assemblies in the cask storage pit and after permanent plant shutdown 294 fuel assemblies in the refueling canal. The heat load from spent fuel stored in the refueling canal racks shall not exceed 1.72x10E6 BTU/Hr. Fuel shall not be stored in the spent fuel racks in the cask storage pit or the refueling canal unless all of the racks are installed in each respective area per the design.

5.7 NOT USED

Item A

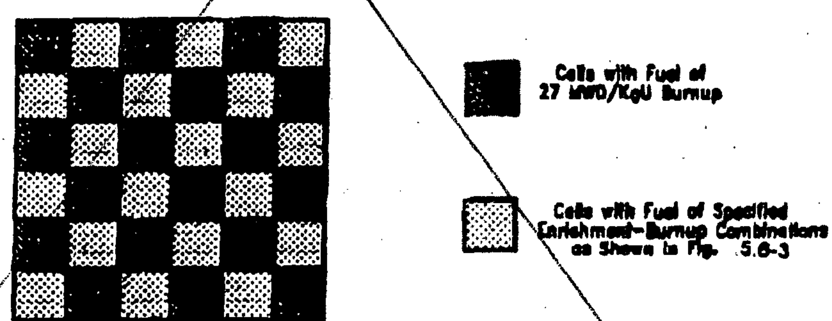
For Region 1 (cask storage pit) and Region 2 (spent fuel pool and refueling canal) racks, a maximum k_{eff} of less than 1.00 when flooded with unborated water, and less than, or equal to, 0.95 when flooded with water having a boron concentration of 447 ppm.

REPLACE WITH INSERT 1



Checkerboard Arrangement of Fresh Assemblies and Empty Cells

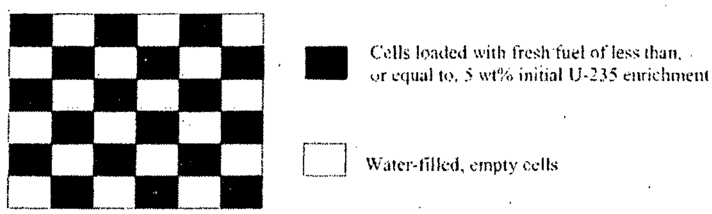
Note: Either of these Checkerboard Arrangements may be used in areas contiguous to each other or to areas of unrestricted storage in Region 2.



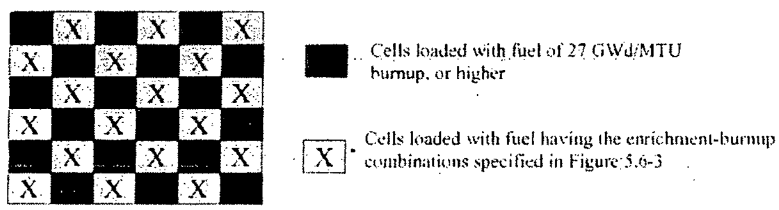
Checkerboard of Fuel Assemblies Burned to 27 MWD/KgU and Fuel of Specified Enrichment-Burnup Combinations

Figure 5.6-1 Alternative Checkerboard Arrangements

INSERT 1



Checkerboard Arrangement of Fresh Fuel Assemblies and Empty Storage Cells



Checkerboard of Fuel Assemblies with Burnups of 27 GWd/MTU, or higher, and Fuel Assemblies of Specified Enrichment-Burnup Combinations

Note: Either of these checkerboard arrangements may be used in areas contiguous to areas of unrestricted storage in Region 2. For interfaces between a fresh fuel checkerboard and an irradiated fuel checkerboard, each high-reactivity irradiated assembly (e.g., 27 GWd/MTU) may be face-adjacent to no more than one fresh fuel assembly; each fresh fuel assembly may be face-adjacent with up to two high-reactivity irradiated fuel assemblies. See Figure 5.6-4 for examples of contiguous fresh fuel checkerboards and irradiated fuel checkerboards which meet these requirements.

Figure 5.6-1 Alternative Checkerboard Storage Arrangements

REPLACE WITH INSERT 2

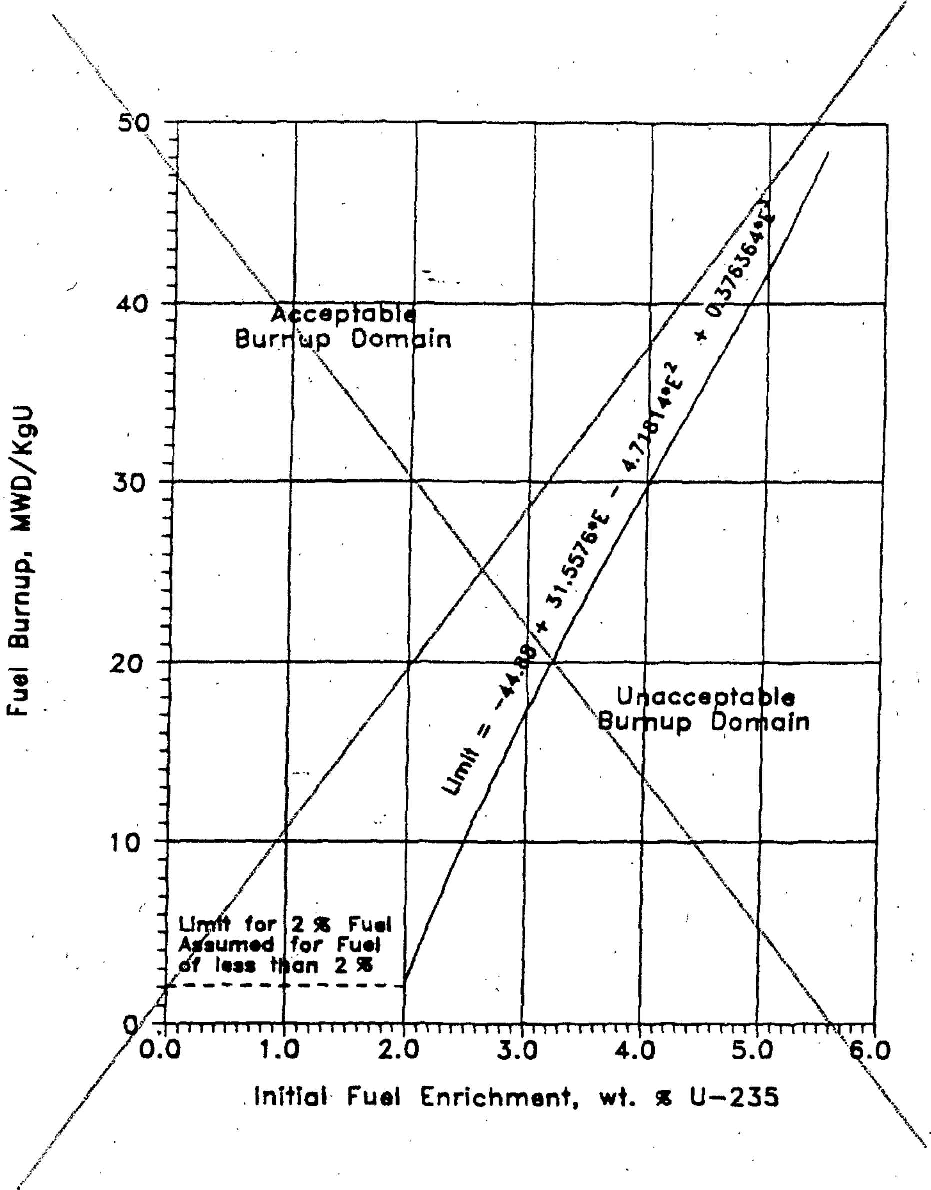


Figure 5.6-2 Acceptable Burnup Domain for Unrestricted Storage of Spent Fuel in Region 2

INSERT 2

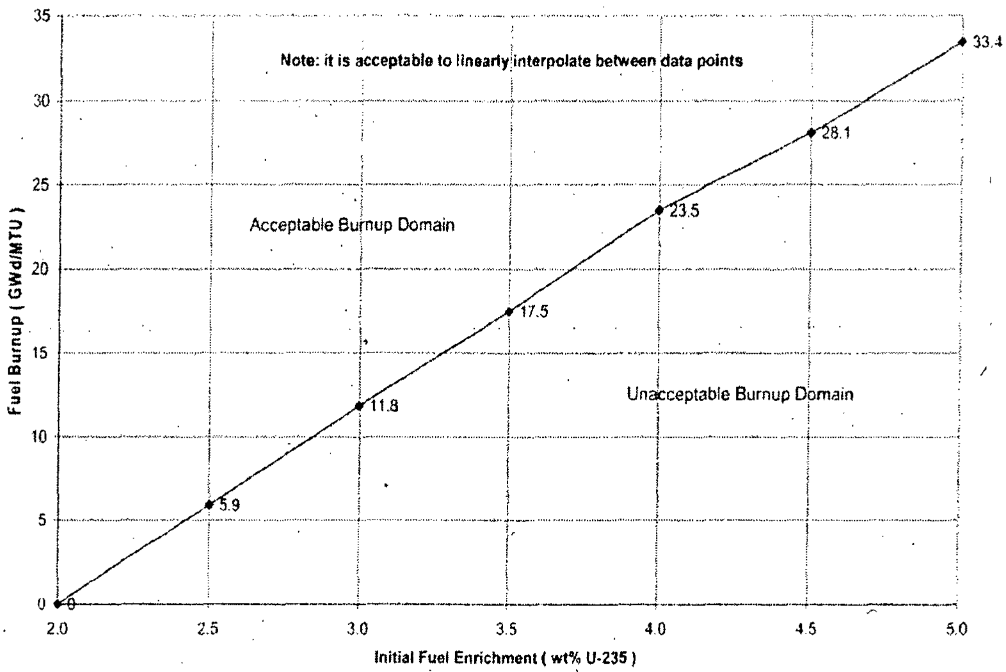


Figure 5.6-2 Acceptable Burnup Domain for Unrestricted Storage of Irradiated Fuel in Region 2 of the Spent Fuel Pool

REPLACE WITH INSERTS 3 AND 4

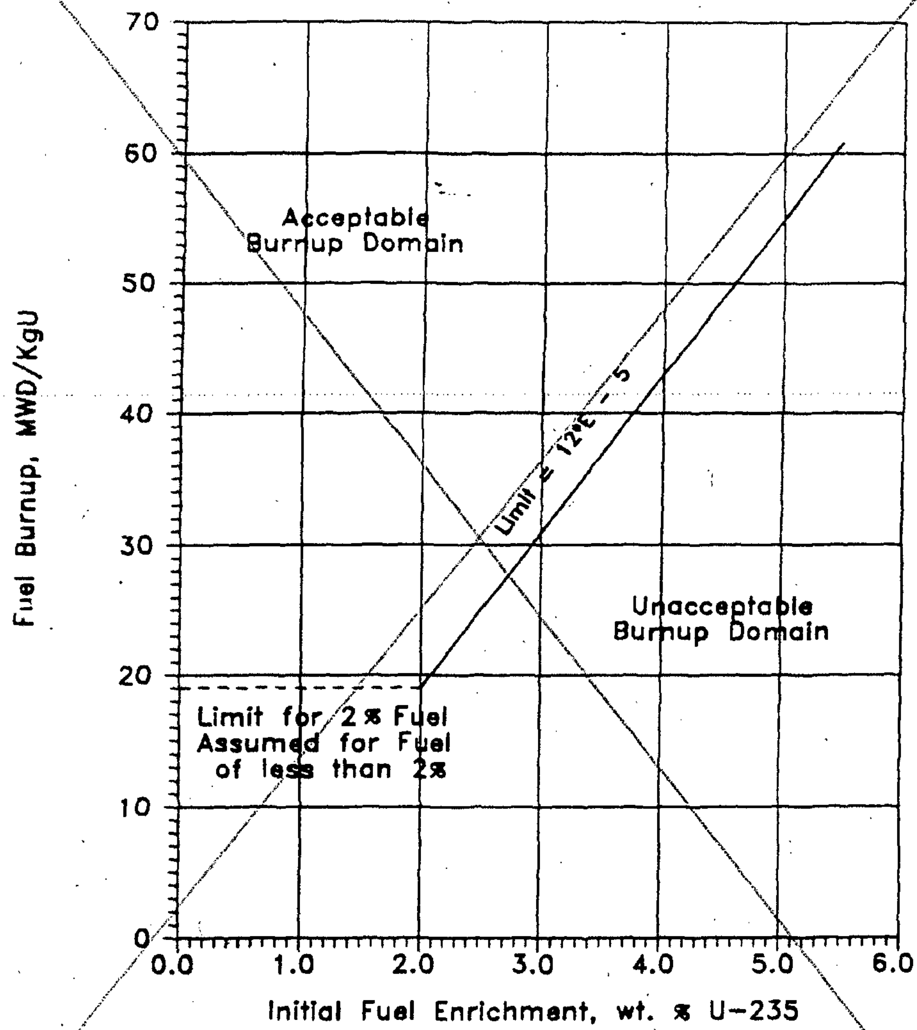


Figure 5.6-3 Acceptable Burnup Domain for Spent Fuel in Checkerboard Arrangement with Fuel of 5 % Enrichment (or less) at 27 MWD/KgU

INSERT 3

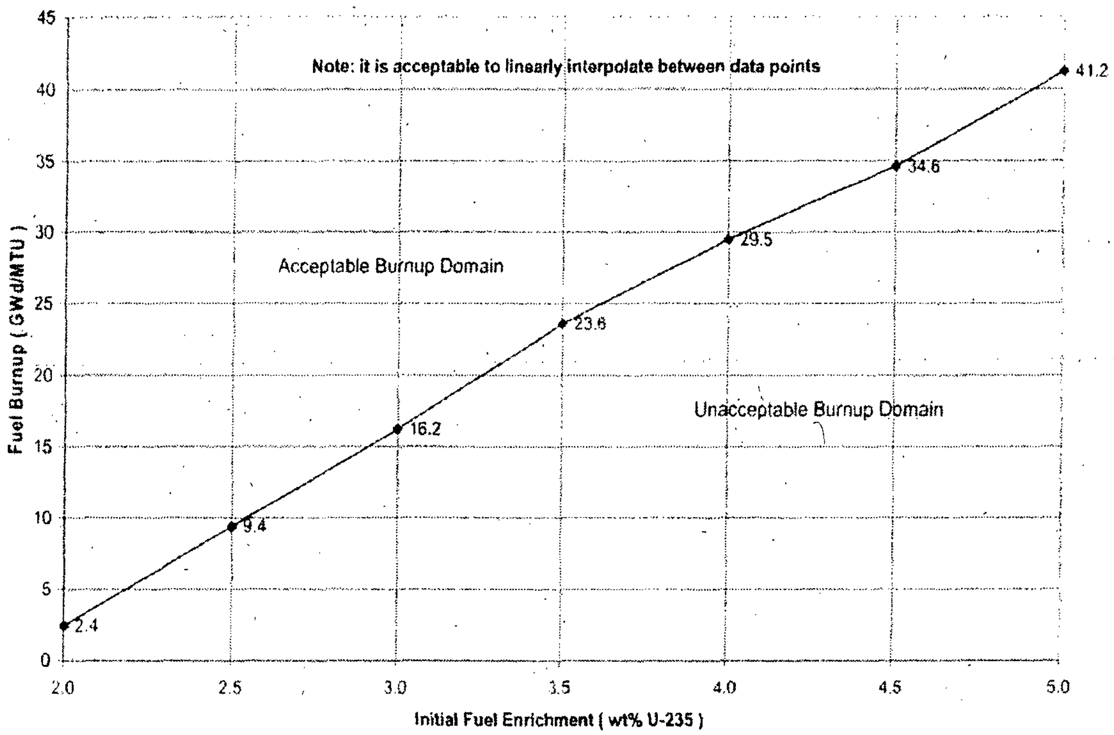


Figure 5.6-3 Acceptable Burnup Domain for Irradiated Fuel in a Checkerboard Arrangement with Fuel of 5 wt% Enrichment, or Less, at 27 GWd/MTU Burnup, or Higher, in Region 2 of the Spent Fuel Pool

INSERT 4

A		A		A		A
	A		A		A	
A		B	C	B		A
	A	C	B	C	A	
A		B	C	B		A
	A		A		A	
A		A		A		A

	≤ 5 wt% U-235, ≥ 27 GWd/MTU
A	Irradiated fuel at, or above, checkerboard curve
B	≤ 5 wt% U-235 fresh fuel
C	Empty storage cell

	A		B	C	B	C
A		A	C	B	C	B
	A		B	C	B	C
A		A	C	B	C	B
	A		B	C	B	C
A		A	C	B	C	B
	A		B	C	B	C

	≤ 5 wt% U-235, ≥ 27 GWd/MTU
A	Irradiated fuel at, or above, checkerboard curve
B	≤ 5 wt% U-235 fresh fuel
C	Empty storage cell

Figure 5.6-4 Examples of Contiguous Fresh Fuel and Irradiated Fuel Checkerboards Which Meet Interface Requirements

ADD TS 3 / 4 9.12

3/4 .9.12 SPENT FUEL POOL (SFP) BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.12 The spent fuel pool boron concentration shall be $\geq$ 1900 ppm.

APPLICABILITY: When fuel assemblies are stored in the SFP

ACTION:

a. With the spent fuel pool boron concentration not within limits immediately suspend movement of fuel in the SFP and immediately initiate actions to restore boron concentration to within limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 Verify the spent fuel pool concentration is within limits once per 7 days.

Attachment 4

W3F1-2008-0052

Affidavit for Proprietary Information

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk

AFFIDAVIT PURSUANT TO 10 CFR 2.390

I, Evan Rosenbaum, being duly sworn, depose and state as follows:

(1) I am the Holtec International Wet Storage Technical Lead for the Waterford Steam Electric Station (WSES) Unit 3 Criticality Analysis Project and have reviewed the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.

(2) The information sought to be withheld is Revision 0 of Holtec Report HI-2084014 containing Holtec Proprietary information.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission</u>, 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA</u>, 704F2d1280 (DC Cir. 1983).

AFFIDAVIT PURSUANT TO 10 CFR 2.390

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a and 4.b, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of

AFFIDAVIT PURSUANT TO 10 CFR 2.390

a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical

AFFIDAVIT PURSUANT TO 10 CFR 2.390

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approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk

AFFIDAVIT PURSUANT TO 10 CFR 2.390

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Mr. Evan Rosenbaum, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 20th day of August, 2008.

Evan Rosenbaum

Evan Rosenbaum, P.E.
Holtec International

Subscribed and sworn before me this 20th day of August, 2008.

Maria C Massi

MARIA C. MASSI
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2010

Attachment 5

W3F1-2008-0052

Licensing Report for W3 NGF Criticality Analysis
(Non Proprietary Information)

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797- 0900
Fax (856) 797 - 0909

LICENSING REPORT FOR WATERFORD UNIT 3 SPENT FUEL POOL CRITICALITY ANALYSIS

FOR

ENTERGY

Holtec Report No: HI-2084014

Holtec Project No: 1712

Report Class : SAFETY RELATED

HOLTEC INTERNATIONAL

DOCUMENT ISSUANCE AND REVISION STATUS[1]

DOCUMENT NAME:	LICENSING REPORT FOR WATERFORD UNIT 3 SPENT FUEL POOL CRITICALITY ANALYSIS				
DOCUMENT NO.:	HI-2084014		CATEGORY: ☐ GENERIC		
PROJECT NO.:	1712		☑ PROJECT SPECIFIC		

Rev. No.[2]	Date Approved	Author's Initials	VIR #
0	7/17/2008	B.Brickner	208892

DOCUMENT CATEGORIZATION

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

☐ Calculation Package[3] (Per HQP 3.2)

☑ 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

The formatting of the contents of this document is in accordance with the instructions of HQP 3.2 or 3.4 except as noted below:

DECLARATION OF PROPRIETARY STATUS

☑ Nonproprietary ☐ Holtec Proprietary ☐ Privileged Intellectual Property (PIP)

Notes

1. This document has been subjected to review, verification and approval process set forth in the Holtec Quality Assurance Procedures Manual. Password controlled signatures of Holtec personnel who participated in the preparation, review, and QA validation of this document are saved on the company's network. The Validation Identifier Record (VIR) number is a random number that is generated by the computer after the specific revision of this document has undergone the required review and approval process, and the appropriate Holtec personnel have recorded their password-controlled electronic concurrence to the document.

2. A revision to this document will be ordered by the Project Manager and carried out if any of its contents is materially affected during evolution of this project. The determination as to the need for revision will be made by the Project Manager with input from others, as deemed necessary by him.

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

Table of Contents

1. INTRODUCTION

This report documents the criticality safety evaluation for the storage of Standard and Next Generation Fuel (NGF) assemblies in Holtec Region 1 & 2 style high-density spent fuel storage racks (SFSRs) at the Waterford Unit 3 nuclear power plant operated by Entergy Nuclear. The purpose of the present analysis is to re-perform the original criticality analysis, taking credit for soluble boron, in order to qualify the racks, etc. for the storage and handling of fuel assemblies having new fuel parameters.

Additional calculations are also documented such as the criticality analysis for storing fuel with an initial enrichment of up to 5.0 wt% ^{235}U in the Reactor Building Temporary Storage Rack (TSR) and storing fuel rods with an initial enrichment of up to 5.0 wt% ^{235}U in the Fuel Pin Storage Container (FPSC) in the spent fuel pool, a boron dilution analysis of the spent fuel pool, a criticality analysis of additional spent fuel pool equipment and also the New Fuel Storage Vault (NFV) (See Section 5.6).

The results of the Region 1 calculations are summarized in Table 7.1 through Table 7.6. The calculations demonstrate that maximum k_{eff} is less than 1.0 without credit for soluble boron and less than or equal to 0.95 with 61 ppm soluble boron. Furthermore, all 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 with 170 ppm soluble boron present.

The results of the Region 2 calculations are summarized in Table 7.7 through Table 7.22, and Table 7.26 through Table 7.27. Under normal conditions, a soluble boron concentration of 447 ppm is required in the spent fuel pool. Under credible accident conditions, a soluble boron concentration of 838 ppm is required (see Table 7.21).

Three loading patterns have been qualified for the Region 2 racks (See Tables 7.16 through Table 7.20):

- a uniform loading of spent fuel meeting the burnup versus enrichment requirements of Table 7.26 ,
- a checkerboard of high and low reactivity fuel (i.e., spent fuel checkerboard). The high reactivity fuel assembly must have an enrichment no greater than 5.0 wt% ^{235}U and a burnup greater than 27 GWD/MTU and the low reactivity fuel must meet the burnup versus enrichment requirements of Table 7.27,
- a checkerboard of fresh fuel up to 5.0 wt% ^{235}U and empty cell locations (i.e., fresh fuel checkerboard).

Within Region 2 racks, several interfaces are possible with the three loading patterns qualified for storage. The permissible interface conditions are summarized as follows:

- No restrictions are necessary between the uniform loading pattern and either of the checkerboard loading patterns (fresh or spent).

- For interfaces between a fresh fuel checkerboard and spent fuel checkerboard, the high reactivity spent fuel assembly (5.0 wt% ^{235}U, 27 GWD/MTU) may be face adjacent to no more than one fresh fuel assembly. The fresh fuel assembly may be face adjacent with up to 2 high reactivity spent fuel assemblies. Figure 7.4 shows one example of an acceptable 3x3 fresh fuel checkerboard within the center of a spent fuel checkerboard that meets these requirements.

2. METHODOLOGY

2.1 Criticality Analysis

The principal method for the criticality analysis of the high-density storage racks is the use of the three-dimensional Monte Carlo code MCNP4a [2]. MCNP4a is a continuous energy three-dimensional Monte Carlo code developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been used previously and verified for criticality analyses and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data predominantly based on ENDF/B-V and ENDF/B-VI. Exceptions are two lumped fission products calculated by the CASMO-4 depletion code, which do not have corresponding cross sections in MCNP4a. For these isotopes, the CASMO-4 cross sections are used in MCNP4a. This approach has been validated in [3] by showing that the cross sections result in the same reactivity effect in both CASMO-4 and MCNP4a.

Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009 with an uncertainty of ± 0.0011 for MCNP4a, evaluated with a 95% probability at the 95% confidence level [1]. The calculations for this analysis utilize the same computer platform and cross-section libraries used for the benchmark calculations discussed in Appendix A.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information has been used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in storage rack criticality calculations. Based on these studies, a minimum of 10,000 histories were simulated per cycle, a minimum of 50 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was usually specified as uniform over the fueled regions (assemblies). Further, the output was reviewed to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between calculational precision and computational time.

Fuel depletion analyses during core operation were performed with CASMO-4 (using the 70-group cross-section library), a two-dimensional multigroup transport theory code based on the Method of Characteristics [4-6]. Detailed neutron energy spectra for each rod type are obtained in collision probability micro-group calculations for use in the condensation of the cross sections. CASMO-4 is used to determine the isotopic composition of the spent fuel. In addition, the CASMO-4 calculations are restarted in the storage rack geometry, yielding the two-dimensional infinite multiplication factor (k_{inf}) for the storage rack to determine the reactivity effect of fuel and rack tolerances, temperature variation, and to perform various studies. For all calculations in the spent fuel pool racks, the Xe-135 concentration in the fuel is conservatively set to zero.

The maximum k_{eff} is determined from the MCNP4a calculated k_{eff}, the calculational bias, the temperature bias, and the applicable uncertainties and tolerances (bias uncertainty, calculational uncertainty, rack tolerances, fuel tolerances, depletion uncertainty) using the following formula:

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

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, except for the assessment of certain accident conditions.

2.2 Boron Dilution Accident

The methodology related to the Boron Dilution accident follows the general equation for boron dilution which is,

$$C_t = C_o e^{-\frac{F}{V}t},$$

where

C_t	=	boron concentration at time t,
C_o	=	initial boron concentration,
V	=	volume of water in the pool, and
F	=	flow rate of un-borated water into the pool

This equation conservatively assumes the un-borated water flowing into the pool mixes instantaneously with the water in the pool.

For convenience, the above equation may be re-arranged to permit calculating the time required to dilute the soluble boron from its initial concentration to a specified minimum concentration, which is given below.

$$t = \frac{V}{F}\ln(C_o / C_t)$$

If V is expressed in gallons and F in gallons per minute (gpm), the time, t, will be in minutes.

3. ACCEPTANCE CRITERIA

The high-density spent fuel PWR storage racks for Waterford Unit 3 are designed in accordance with the applicable codes and standards listed below. The objective of this evaluation is to show that the effective neutron multiplication factor, k_{eff}, is less than 1.0 with the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with un-borated water at a temperature corresponding to the highest reactivity. In addition, it is to be demonstrated that k_{eff} is less than or equal to 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with borated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations including manufacturing tolerances and is shown to be less than 0.95 with a 95% probability at a 95% confidence level [1]. 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 under borated conditions.

Applicable codes, standard, 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).
- L. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2, March 2007.
- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- Code of Federal Regulations, Title 10, Part 50, Section 68, "Criticality Accident Requirements."

The New Fuel Storage Vault is intended for the receipt and storage of fresh fuel under normally dry conditions where the reactivity is very low. To assure criticality safety under accident conditions and to conform to the requirements of 10 CFR 50.68, these two accident condition criteria must be met:

- When fully loaded with fuel of the highest anticipated reactivity and flooded with clean unborated water, the maximum reactivity, including uncertainties, shall not exceed a k_{eff} of 0.95.

- With fuel of the highest anticipated reactivity in place and assuming the optimum hypothetical low density moderation, (i.e., fog or foam), the maximum reactivity shall not exceed a k_{eff} of 0.98.

These criteria preclude a secondary accident per ANSI 8.1 or accidents under dry conditions.

4. ASSUMPTIONS

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were employed:

1) Moderator is borated or un-borated water at a temperature in the operating range that results in the highest reactivity, as determined by the analysis.

2) Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water.

3) The effective multiplication factor of an infinite radial array of fuel assemblies was used in the analyses, except for the assessment of certain abnormal/accident conditions and conditions where leakage is inherent.

4) The neutron absorber length is modeled to be the same length as the active region of the fuel.

5) No cooling time is credited in the rack calculations.

6) The presence of burnable absorbers in fresh fuel is neglected. This is conservative as burnable absorbers would reduce the reactivity of the fresh fuel assembly.

7) The presence of annular pellets is neglected. This is conservative as it is bounded by the solid fuel.

8) All structural materials of the new fuel storage racks are conservatively neglected and replaced with water at the appropriate density.

9) The concrete wall of the transfer canal is conservatively modeled as 100 cm thick.

10) The FPSC tubes holes were not modeled; however, the other steel structures of the FPSC were modeled as water. Therefore, the neglecting of the tube holes is conservative.

11) The concrete walls of the vault are conservatively modeled as 100 cm thick.

12) The two inch redwood planks in the NFV are assumed to be 1.5 inches thick.

13) In MCNP4a, the Doppler treatment and cross-sections are valid at 300K (80.33 °F); however, in the NFV calculations no temperature bias is applied to the results to account for the actual temperature of the water.

14) In the NFV the eccentric fuel positioning condition is covered by the fuel cell spacing tolerance.

5. INPUT DATA

5.1 Fuel Assembly Specification

The spent fuel storage racks are designed to accommodate various 16x16 fuel assemblies used at the Waterford Unit 3 facility. The design specifications for these fuel assemblies, which were used for this analysis, are given in Table 5.1.

5.2 Core Operating Parameters

Core operating parameters are necessary for fuel depletion calculations performed with CASMO-4. The core parameters used for the depletion calculations are presented in Table 5.2. Temperature and soluble boron values are taken as the upper bound (most conservative) of the core operating parameters of Waterford Unit 3. The neutron spectrum is hardened by each of these parameters, leading to a greater production of plutonium during depletion, which results in conservative reactivity values.

5.3 Axial Burnup Distribution

Generic axial burnup profiles provided by the client are specified at node centers for 24 equally-spaced axial sections for burnups of less than 25 GWD/MTU and greater than 25 GWD/MTU. The resulting profiles are presented in Table 5.3.

5.4 Burnable Absorbers

At the Waterford Unit 3 facility there is the potential for either B_4C, erbia or IFBA burnable absorbers to be located in the fuel assembly as integral absorbers. In [10] it is clearly seen that the reactivity of the fuel assembly with IFBA bound those with B_4C or erbia and therefore only the IFBA is considered in this analysis. The design specifications for the IFBA rods are given in Table 5.1 and are further discussed in Section 7.2.2.

5.5 Storage Rack Specification

The storage cell characteristics are summarized in Table 5.4.

5.5.1 Region 1 Style Storage Racks

The Region 1 storage cells are composed of stainless steel boxes separated by a water gap, with fixed neutron absorber panels centered on each side. The steel walls define the storage cells, and stainless steel sheathing supports the neutron absorber panel and defines the boundary of the flux-trap water-gap used to augment reactivity control. Stainless steel channels connect the storage cells in a rigid structure and define the flux-trap between the neutron absorber panels. Neutron absorber panels are installed on all exterior walls facing other racks.

The calculational models consist of a single cell with reflective boundary conditions through the centerline of the water gaps, thus simulating an infinite array of Region 1 storage cells. Figure 5.1 shows the actual calculational model containing the reference 16x16 assembly, as drawn by the two-dimensional plotter in MCNP4a. The calculations are described in Section 7.1.

5.5.2 Region 2 Style Storage Racks

The Region 2 storage cells are composed of stainless steel boxes with a single fixed neutron absorber panel, (attached by stainless steel sheathing) centered on each side. The stainless steel boxes are arranged in an alternating pattern such that the connection of the box corners form storage cells between those of the stainless steel boxes.

The calculational models consist of a group of four identical cells surrounded by reflective boundary conditions through the centerline of the composite of materials between the cells, thus simulating an infinite array of Region 2 storage cells. Figure 5.2 shows the actual calculational model containing the 16x16 assembly as drawn by the two-dimensional plotter in MCNP4a.

5.5.3 Rack Interfaces

Based on the layout of the spent fuel pool, there are no Region 1 to Region 2 interfaces. The gap between adjacent Region 2 racks is conservatively neglected. The Region 2 to Region 2 rack loading pattern interfaces are analyzed in Section 7.3.

5.6 Additional Calculations

5.6.1 Fuel Transfer Carriage Criticality

The fuel transfer carriage conveys the fuel assemblies through the fuel transfer tube and is capable of accommodating two fuel assemblies at a time, carried in stainless steel boxes. The results of this calculation can be found in Section 7.4.1.

5.6.2 Upender Criticality

The fuel upender is a machine located at each end of the transfer tube. The criticality of this component is bounded by the fuel transfer carriage. No input required. See Section 7.4.2.

5.6.3 New Fuel Elevator Criticality

The new fuel elevator has a capacity of a single fuel assembly and is utilized to lower new fuel from the operating level of the fuel handling building to the bottom of the spent fuel pool. See Section 7.4.3.

5.6.4 Boron Dilution Accident Evaluation

The spent fuel pool at Waterford Unit 3 has a minimum soluble boron concentration of 1720 ppm. The spent fuel pool volume is considered to be 38,600 ft^3. Under certain abnormal conditions, un-borated water may dilute this concentration below the requirements determined in Section 7.

Makeup to the spent fuel storage pool is from the Refueling Water Storage Pool and/or the Condensate Storage Pool. Makeup from the Refueling Water Storage Pool is provided by the refueling water pool purification pump which has a capacity of 150 gpm. The Refueling Water Storage Pool has a minimum boron concentration of 2050 ppm. The component cooling water makeup pumps provide makeup from the Condensate Storage Pool and have a capacity of 600 gpm. For the accident case a high flow rate of 600 gpm is therefore assumed. The results of these calculations are shown in Section 7.4.4.

5.6.5 Temporary Storage Racks

The TSR storage cell locations are arranged in a row of 5 cells with the geometric dimensions in Table 5.5. The design basis calculational model places 5 fresh fuel assemblies enriched to 5.0 wt% ^{235}U in the storage rack. No steel structural material is included. For simplification, the following tolerances are included in the design basis model: fuel density, lattice pitch and enrichment.

5.6.6 Fuel Pin Storage Container

The FPSC is a square stainless steel container that fits in a fuel assembly storage rack in the spent fuel pool. It has 81 stainless steel tubes that may contain fuel rods of up to 5.0 wt% ^{235}U (See Table 5.5). The FPSC was modeled as 81 solid steel tubes of equal diameter, each containing 1 fresh fuel rod with the maximum enrichment. All other steel components of the container were neglected. The model includes 100 cm of water surrounding the FPSC or fuel assembly.

The criticality analysis of the FRSC is performed by comparing the reactivity of the FRSC loaded with the maximum number of fresh fuel pins to the reactivity of various fuel assemblies and determine which cases bound the FRSC. These calculations are performed with the fuel assembly surrounded by 100 cm of water, meaning no storage racks, poison material or structural materials are considered (the steel tubes of the FRSC are modeled). No tolerances are included. Reflective boundary conditions are applied on all sides to maximize reactivity.

5.6.7 New Fuel Storage Vault

The NGF assembly is the only fuel assembly type to be stored in the NFV. The design input data is tabulated in Table 5.1 and Table 5.6. The storage locations are arranged in 8 modules providing a total of 16 rows of 5 cells each for a total of 80 storage locations. The cells are located on a 21 inch pitch within each module, and on a 49 inch cell center to cell center spacing between modules in the east-west direction and a 58 inch cell center to cell center spacing between modules in the north-south direction. Normally, fuel is stored in the dry condition with very low reactivity. Graphic representations of the analytical model are shown in Figure 7.5 and 7.6. These figures were drawn (to scale) with a two-dimensional plotter.

The reactivity uncertainties associated with various manufacturing tolerances for the NFV were calculated by the difference between two MCNP4a calculations, one with the nominal value and a second independent calculation with the tolerance parameter changed. Based on the nominal condition results, it was determined that the 100% moderator condition, i.e. 1.0 g/cc, represented the maximum reactivity condition and therefore the tolerance calculations were performed with 100% moderator density. These tolerance effects each include the combination of statistical errors in the MCNP4a calculations due to the random nature of Monte Carlo calculations, at the 95% confidence level ($\Delta k+(\sqrt{2})*2*\sigma$). In evaluating the uncertainties due to tolerances, the following tolerances were used:

- Enrichment Tolerance of ± 0.05 wt% ^{235}U
- Density of ±0.165 g UO_2/cm^3
- Fuel Storage Cell Spacing of ±0.8125 in.

The fuel storage cell spacing tolerance was only used in the 21 inch assembly pitch. In determining the maximum k_{eff}, the effects of these manufacturing tolerances were statistically combined (square root of the sum of the squares) with the MCNP4a bias uncertainty from the benchmarking results and the MCNP4a calculational statistics ($2*\sigma$) to determine the total uncertainty.

6. COMPUTER CODES

The following computer codes were used during this analysis.

- MCNP4a [2] is a three-dimensional continuous energy Monte Carlo code developed at Los Alamos National Laboratory. This code offers the capability of performing full three-dimensional calculations for the loaded storage racks. MCNP4a was run on the PCs at Holtec.

- CASMO-4, Version 2.05.14 [4-6] is a two-dimensional multigroup transport theory code developed by Studsvik Scandpower, Inc. CASMO-4 performs cell criticality calculations and burnup. CASMO-4 has the capability of analytically restarting burned fuel assemblies in the rack configuration. This code was used to determine the reactivity effects of tolerances and fuel depletion.

7. ANALYSIS

This section describes the calculations that were used to determine the acceptable storage criteria for the Region 1 and Region 2 style racks. In addition, this section discusses the possible abnormal and accident conditions.

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 discussed below.

As discussed in Section 2, MCNP4a was the primary code used in the PWR calculations. CASMO-4 was used to determine the reactivity effect of tolerances and for depletion calculations. MCNP4a was used for reference cases and to perform calculations which are not possible with CASMO-4 (e.g., eccentric fuel positioning, axial burnup distributions, and fuel misloading).

Figures 5.1 and 5.2 are pictures of the basic calculational models used in MCNP4a. These pictures were created with the two-dimensional plotter in MCNP4a and clearly indicate the explicit modeling of fuel rods in each fuel assembly. In CASMO-4, a single cell is modeled, and since CASMO-4 is a two-dimensional code, the fuel assembly hardware above and below the active fuel length is not represented. The three-dimensional MCNP4a models that included axial leakage assumed approximately 30 cm of water above and below the active fuel length. Additional models with more storage cells were generated with MCNP4a to investigate the effect of abnormal and normal conditions. These models are discussed in the appropriate section.

7.1 Region 1

The goal of the criticality calculations for the Region 1 style racks is to qualify the racks for storage of fuel assemblies with design specifications as shown in Table 5.1 and a maximum nominal initial enrichment of 5.0 wt% ^{235}U.

7.1.1 Identification of Reference Fuel Assembly

CASMO-4 calculations were performed to determine which of the two assembly types in Table 5.1 is bounding in the Region 1 racks. The presence of burnable absorbers in the fuel assembly (IFBA) was neglected for determination of the reference fuel assembly. The results in Table 7.1

shows that the NGF assembly has the highest reactivity and this assembly type is therefore used in all subsequent calculations.

7.1.2 Eccentric Fuel Assembly Positioning

The fuel assemblies are assumed to be normally located in the center of the storage rack cell. To investigate the potential reactivity effect of eccentric positioning of assemblies in the cells, MCNP4a calculations were performed with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). The highest reactivity, therefore, corresponds to the reference design with the fuel assemblies positioned in the center of the storage cells. The results of this calculation is shown in Table 7.6.

7.1.3 Uncertainties Due to Manufacturing Tolerances

In the calculation of the final k_{eff}, the effect of manufacturing tolerances on reactivity must be included. CASMO-4 was used to perform these calculations. As allowed in [7], the methodology employed to calculate the tolerance effects combine both the worst-case bounding value and sensitivity study approaches. The evaluations include tolerances of the rack and fuel dimensions. As for the bounding assembly, calculations are performed at an enrichment of 5.0 wt% ^{235}U. The reference condition is the condition with nominal dimensions and properties. To determine the Δk associated with a specific manufacturing tolerance, the k_{inf} calculated for the reference condition is compared to the k_{inf} from a calculation with 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 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. The fuel and rack tolerances included in this analysis are described below; the fuel density and enrichment tolerances are typical values:

Fuel Tolerances

- Increased Fuel Density: +0.165 g/cm^3
- Increased Fuel Enrichment: 0.05 wt% ^{235}U
- Fuel Rod Pitch: ±0.01 in.
- Fuel Rod Cladding Outside Diameter: ± 0.0015 in.
- Fuel Rod Cladding Thickness min: 0.021 in.
- Fuel Pellet Outside Diameter: ± 0.0005 in.
- Guide Tube Outside Diameter: ± 0.003 in.
- Guide Tube Thickness min: 0.036 in.

Rack Tolerances

- Cell Inner Dimension: [redacted]
- Box Wall Thickness: [redacted]
- Cell Pitch: [redacted]
- Boral Width: [redacted]

- Poison Gap min: [redacted]
- Poison Loading min: [redacted]

Regarding the tolerance calculations, the following needs to be noted:

- In some cases it is not obvious whether an increase or decrease of the parameter will lead to an increase in reactivity. In these cases, the reactivity effect of both increase and decrease of the parameter are calculated, and the positive reactivity effect is used when calculating the statistical combination.

- The tolerance in the flux trap is conservatively captured in the tolerances of the cell ID and cell pitch, since variations of the cell ID are evaluated for a constant cell pitch and vice versa.

- Tolerance calculations were performed for pure water only since the presence of soluble boron in the pool lowers reactivity and reactivity effects of tolerances, and therefore the pure water case bounds the soluble boron case.

The results of the calculations of the manufacturing tolerances are presented in Table 7.2.

7.1.4 Temperature and Water Density Effects

Pool water temperature effects on reactivity in the Region 1 racks have been calculated with CASMO-4 for various enrichments with a maximum value of 5.0 wt% ^{235}U and the results are presented in Table 7.3. The results show that the Region 1 spent fuel pool temperature coefficient of reactivity is negative, i.e., a lower temperature results in a higher reactivity. Consequently, the design basis calculations are evaluated at 0 °C (32 °F) for normal conditions.

In MCNP4a, the Doppler treatment and cross-sections are valid only at 300K (80.33 °F). Therefore, a Δk is determined in CASMO-4 from 32 °F to 80.33 °F, and is included in the final k_{eff} calculation as a bias. Table 7.3 shows the calculation of the bias. The temperature bias is calculated with pure water.

7.1.5 Calculation of Maximum k_{eff}

Using the calculational model shown in Figure 5.1 and the reference 16x16 NGF fuel assemblies, the k_{eff} in the Region 1 storage racks has been calculated with MCNP4a. The calculations of the maximum k_{eff} values, based on the formula in Section 2, are shown in Table 7.4 and Table 7.5. In summary, the results show that the maximum k_{eff} of the Region 1 racks is less than 1.0 at a 95% probability at a 95% confidence level with no credit for soluble boron, and by linear interpolation, less than or equal to 0.95 with 61 ppm soluble boron.

7.1.6 Abnormal and Accident Conditions

The effects on reactivity of credible abnormal and accident conditions are examined in this section. This section identifies which of the credible abnormal or accident conditions will result in exceeding the limiting reactivity ($k_{eff} \leq 0.95$). For those accident or abnormal conditions that result in exceeding the limiting reactivity, a minimum soluble boron concentration is determined to ensure that $k_{eff} \leq 0.95$. The double contingency principal of ANS-8.1/N16.1-1975 [8] (and the USNRC letter of April 1978; see Section 3.0) specifies that it shall require at least two unlikely, independent and concurrent events to produce a criticality accident. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions.

7.1.6.1 Abnormal Temperature

All calculations for Region 1 are performed at a pool temperature of 32°F. As shown in Section 7.1.4 above, the temperature coefficient of reactivity is negative, therefore any increase in temperature above 32°F would cause a reduction in the reactivity. Therefore, no further evaluations of abnormal temperatures are performed.

7.1.6.2 Dropped Assembly - Horizontal

For the case in which a fuel assembly is assumed to be dropped on top of a 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 than 12 inches, which is sufficient to preclude neutron coupling (i.e., an effectively infinite separation). Consequently, the horizontal fuel assembly drop accident will not result in a significant increase in reactivity. Furthermore, the soluble boron in the spent fuel pool water assures that the true reactivity is always less than the limiting value for this dropped fuel accident.

7.1.6.3 Dropped Assembly – Vertical Into Fuel Cell

It is also possible to vertically drop an assembly into a location that might be occupied by another assembly or that might be empty. Such a vertical impact onto another assembly has previously been shown to cause no damage to either fuel assembly. A vertical drop into an empty storage cell could result in a small deformation of the baseplate. The resultant effect would be the lowering of a single fuel assembly by the amount of the deformation. This could potentially result in further misalignment between the active fuel region and the Boral. However, the amount of deformation for this drop would be small and restricted to a localized area of the rack around the storage cell where the drop occurs. Furthermore, the soluble boron in the spent fuel pool water assures that the true reactivity is always less than the limiting value for this dropped fuel accident.

7.1.6.4 Abnormal Location of a Fuel Assembly

7.1.6.4.1 Misloaded Fresh Fuel Assembly

The Region 1 racks are qualified for the storage of fresh, unburned fuel assemblies with the maximum permissible enrichment (5.0 wt% ^{235}U). Therefore, the abnormal location of a fuel assembly within normal Region 1 cells is of no concern.

7.1.6.4.2 Mislocated Fresh Fuel Assembly

The mislocation of a fresh unburned fuel assembly could, in the absence of soluble poison, result in exceeding the regulatory limit (k_{eff} of 0.95). This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (5.0 wt% ^{235}U) were to be accidentally mislocated outside of a storage rack adjacent to other fuel assemblies. The results of the analysis are shown in Table 7.6 and show by linear interpolation that a soluble boron level of 170 ppm is sufficient to ensure that the maximum k_{eff} value for this condition remains at or below 0.95

7.2 Region 2

The goal of the criticality calculations for the Region 2 style racks is to qualify the racks for storage of fuel assemblies with design specifications as shown in Table 5.1 and a maximum nominal initial enrichment of 5.0 wt% ^{235}U. Specifically, the purpose of the criticality calculations is to determine the initial enrichment and burnup combinations required for the storage of spent fuel assemblies with nominal initial enrichments up to 5.0 wt% ^{235}U. Three loading configurations were analyzed to create burnup versus enrichment curves:

- a uniform loading of spent fuel meeting the burnup versus enrichment requirements of Table 7.26,
- a checkerboard loading pattern of high and low reactivity fuel with the high reactivity fuel at an enrichment of 5.0 wt% ^{235}U and a burnup of 27 GWD/MTU and the low reactivity fuel must meet the burnup versus enrichment requirements of Table 7.27;
- a checkerboard of fresh fuel up to 5.0 wt% ^{235}U and empty cell locations (i.e., fresh fuel checkerboard).

7.2.1 Identification of Reference Fuel Assembly

CASMO-4 calculations were performed to determine which of the two assembly types are bounding in the Region 2 racks. In the calculations, the fuel assembly is burned in the core configuration and restarted in the rack configuration. For all assemblies, the presence of burnable absorbers in the fuel assembly (BPRA, IFBA) was neglected for determination of the reference fuel assembly (see Section 7.2.2 for a discussion the effect of burnable poison). The results are shown in Table 7.7 (selected enrichments and burnups) and show that the NGF assembly has the highest reactivity for all enrichments and burnups relative to the final burnup versus enrichment curve.

7.2.2 Reactivity Effect of Burnable Absorbers During Depletion

The Waterford Unit 3 fuel makes use of burnable absorbers of either B_4C, erbia or integral fuel burnable absorber (IFBA) rods with a thin coating of ZrB_2 on the UO_2 pellet.

Generic studies [10] have investigated the effect that integral burnable absorbers (IBAs) have on the reactivity of spent fuel assemblies. These studies have concluded that there is a small positive reactivity effect associated with the presence of IFBA rods, which therefore bounds the negative effects of the B_4C and erbia. Therefore, only the IFBA is considered in this analysis. To determine the reactivity effect for the Waterford Unit 3 spent fuel racks, depletion calculations were performed for selected configurations of IFBA rods provided by Entergy. The reactivity of the fuel assembly with IFBA rods is compared to the reactivity of the respective fuel assembly without IFBA rods. The results are presented in Table 7.8 and an IFBA bias of 0.0070 is conservatively applied to the final k_{eff} to bound all IFBA configurations.

7.2.3 Reactivity Effect of Axial Burnup Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower ends. At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of lower reactivity worth due to neutron leakage. Consequently, it would be expected that over most of the burnup history, distributed burnup fuel assemblies would exhibit a slightly lower reactivity than that calculated for the average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup.

Generic analytic results of the axial burnup effect for assemblies without axial blankets have been provided by Turner [9] based upon calculated and measured axial burnup distributions. These analyses confirm the minor and generally negative reactivity effect of the axially distributed burnup compared to a flat distribution, becoming positive at burnups greater than about 30 GWD/MTU. The trends observed in [9] suggest the possibility of a small positive reactivity effect above 30 GWD/MTU, increasing to slightly over 1% Δk at 40 GWD/MTU. The required burnup for the maximum enrichment is higher than 30 GWD/MTU. Therefore, a positive reactivity effect of the axially distributed burnup is possible. Calculations are conservatively performed with the axial burnup distribution shown in Table 5.3 (see Section 5.3) and with an axially constant burnup, and the higher reactivity is used in the analyses.

7.2.4 Isotopic Compositions

To perform the criticality evaluation for spent fuel in MCNP4a, the isotopic composition of the fuel is calculated with the depletion code CASMO-4 and then specified as input data for

MCNP4a. The CASMO-4 calculations performed to obtain the isotopic compositions for MCNP4a were performed generically, with one calculation for each enrichment, and burnups in increments of 2.5 GWD/MTU or less. The isotopic composition for any given burnup is then determined by linear interpolation.

7.2.5 Uncertainty in Depletion Calculations

Since critical experiment data with spent fuel is not available for determining the uncertainty in burnup-dependent reactivity calculations, an allowance for uncertainty in reactivity was assigned based upon other considerations. Based on the recommendation in [7], a burnup dependent uncertainty in reactivity for burnup calculations of 5% of the reactivity decrement is used. This allowance is statistically combined with the other reactivity allowances in the determination of the maximum k_{eff} for normal conditions where assembly burnup is credited.

7.2.6 Eccentric Fuel Assembly Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell. In the absence of a fixed neutron absorber, the eccentric location of fuel assemblies in the storage cells may produce a positive reactivity effect. Therefore, the eccentric positioning is performed in a very conservative manner in MCNP4a, assuming 4 assemblies in the corners of the storage cell (four-assembly cluster at closest approach), and that these clusters of four assemblies are repeated throughout the rack. The results of these calculations are shown in Table 7.9 and indicate that eccentric fuel positioning results in a decrease in reactivity.

7.2.7 Uncertainties Due to Manufacturing Tolerances

In the calculation of the final k_{eff}, the effect of manufacturing tolerances on reactivity must be included. CASMO-4 was used to perform these calculations. As allowed in [7], the methodology employed to calculate the tolerance effects combine both the worst-case bounding value and sensitivity study approaches. The evaluations include tolerances of the rack and fuel dimensions. As for the bounding assembly, calculations are performed for different enrichments and burnups with a maximum value of 5.0 wt% ^{235}U. The reference condition is the condition with nominal dimensions and properties. To determine the Δk associated with a specific manufacturing tolerance, the k_{inf} calculated for the reference condition is compared to the k_{inf} from a calculation with 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 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 fuel and rack tolerances included in this analysis are described below; the fuel density and enrichment tolerances are typical values:

Fuel Tolerances

- Increased Fuel Density: +0.165 g/cm^3
- Increased Fuel Enrichment: 0.05 wt% ^{235}U
- Fuel Rod Pitch: ±0.01 in.
- Fuel Rod Cladding Outside Diameter: ± 0.0015 in.
- Fuel Rod Cladding Thickness min: 0.021 in.
- Fuel Pellet Outside Diameter: ± 0.0005 in.
- Guide Tube Outside Diameter: ± 0.003 in.
- Guide Tube Thickness min: 0.036 in.

Rack Tolerances

- Cell Inner Dimension: [redacted]
- Box Wall Thickness: [redacted]
- Poison Width: [redacted]
- Poison Gap minimum: [redacted]
- Boral B-10 Loading min: [redacted]

Regarding the tolerance calculations, the following needs to be noted:

- In some cases it is not obvious whether an increase or decrease of the parameter will lead to an increase in reactivity. In these cases, the reactivity effect of both increase and decrease of the parameter are calculated, and the positive reactivity effect is used when calculating the statistical combination.

- In the CASMO-4 model used, the tolerance calculation for the Cell ID resulted in a negative reactivity for both increases and decreases in Cell ID. Conservatively, the least negative value was used as a positive reactivity effect.

- Tolerance calculations were performed for pure water only since the presence of soluble boron in the pool lowers reactivity and reactivity effects of tolerances, and therefore the pure water case bounds the soluble boron case.

7.2.8 Temperature and Water Density Effects

Pool water temperature effects on reactivity in the Region 2 racks have been calculated with CASMO-4 for various enrichments with a maximum value of 5.0 wt% ^{235}U and the results are presented in Table 7.12. The results show that the Region 2 spent fuel pool temperature coefficient of reactivity is negative, i.e., a higher temperature results in a lower reactivity. Consequently, all CASMO-4 calculations are evaluated at 32 °F.

In MCNP4a, the Doppler treatment and cross-sections are valid only at 300K (80.33 °F). Therefore, a Δk is determined in CASMO-4 from 32 °F to 80.33 °F, and is included in the final k_{eff} calculation as a bias.

7.2.9 Calculation of Maximum k_{eff}

Using the calculational model shown in Figure 5.2 and the reference 16x16 NGF fuel assembly, the k_{eff} in the Region 2 storage racks has been calculated with MCNP4a for the cases discussed in Section 7.2. The determination of the maximum k_{eff} values, based on the formula in Section 2, is shown in, for initial enrichments between 2.0 wt% ^{235}U and 5.0 wt% ^{235}U, Table 7.13 for the uniform loading case, Table 7.14 for the spent fuel checkerboard loading case, and Table 7.15 for the fresh fuel checkerboard case. A summary of the calculations for non-accident conditions of the maximum k_{eff} for spent fuel of maximum nominal enrichment of 5.0 wt% ^{235}U is shown in Table 7.16 for the uniform loading of spent fuel without soluble boron and Table 7.17 with soluble boron, Table 7.18 for the spent fuel checkerboard without soluble boron and Table 7.19 with soluble boron, and Table 7.20 for the fresh fuel checkerboard fuel. Table 7.26 and Figure 7.1 present the burnup versus enrichment requirements for the uniform loading of spent fuel and Table 7.27 and Figure 7.2 present the burnup versus enrichment requirements for the low reactivity fuel assemblies in the spent fuel checkerboard. The results show that the maximum k_{eff} of the Region 2 racks is less than 1.0 at a 95% probability and at a 95% confidence level for the three loading patterns and less than 0.95 at a 95% probability and at a 95% confidence level with 447 ppm soluble boron.

7.2.10 Abnormal and Accident Conditions

The effects on reactivity of credible abnormal and accident conditions are examined in this section. This section identifies which of the credible abnormal or accident conditions will result in exceeding the limiting reactivity ($k_{eff} \leq 0.95$). For those accident or abnormal conditions that result in exceeding the limiting reactivity, a minimum soluble boron concentration is determined to ensure that $k_{eff} \leq 0.95$. The double contingency principal of ANS-8.1/N16.1-1975 [8] (and the USNRC letter of April 1978; see Section 3.0) specifies that it shall require at least two unlikely, independent and concurrent events to produce a criticality accident. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions.

7.2.10.1 Abnormal Temperature

All calculations for Region 2 are performed at a pool temperature of 32 °F. As shown in Section 7.2.8 above, the temperature coefficient of reactivity is negative, therefore no additional calculations are required.

7.2.10.2 Dropped Assembly - Horizontal

For the case in which a fuel assembly is assumed to be dropped on top of a 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 than 12 inches, which is sufficient to preclude neutron coupling (i.e., an effectively infinite separation). Consequently, the horizontal fuel assembly drop accident will not result in a significant increase in reactivity. Furthermore, the soluble boron in the spent fuel pool water assures that the true reactivity is always less than the limiting value for this dropped fuel accident.

7.2.10.3 Dropped Assembly - Vertical

It is also possible to vertically drop an assembly into a location that might be occupied by another assembly or that might be empty. Such a vertical impact onto another assembly has previously been shown to cause no damage to either fuel assembly. A vertical drop into an empty storage cell could result in a small deformation of the baseplate. The resultant effect would be the lowering of a single fuel assembly by the amount of the deformation. This could potentially result in further misalignment between the active fuel region and the Boral. However, the amount of deformation for this drop would be small and restricted to a localized area of the rack around the storage cell where the drop occurs. Furthermore, the reactivity increase would be small compared to the reactivity increase created by the misloading of a fresh assembly discussed in the following section. The vertical drop is therefore bounded by this misloading accident and no separate calculation is performed for the drop accident.

7.2.10.4 Abnormal Location of a Fuel Assembly

7.2.10.4.1 Misloaded Fresh Fuel Assembly

The misloading of a fresh unburned fuel assembly could, in the absence of soluble poison, result in exceeding the regulatory limit (k_{eff} of 0.95). This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (5.0 wt% ^{235}U) were to be inadvertently misloaded into a storage cell intended to be used for spent fuel. The results of this accident are shown in Table 7.21.

7.2.10.4.2 Mislocated Fresh Fuel Assembly

The mislocation of a fresh unburned fuel assembly could, in the absence of soluble poison, result in exceeding the regulatory limit (k_{eff} of 0.95). This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (5.0 wt% ^{235}U) were to be accidentally mislocated outside of a Region 2 storage rack adjacent to other fuel assemblies

The MCNP4a model consists of an array of Region 2 fuel storage cells with a single fresh, unburned assembly placed adjacent to the rack as close to the rack faces as possible to maximize the possible reactivity effect. The results of the analysis are shown in Table 7.21.

7.3 Interfaces Within and Between Racks

The calculations in Sections 7.1 and 7.2 assume laterally infinite arrangements of rack cells. This section evaluates the potential effect of the interfaces between and within rack modules.

7.3.1 Gaps Between Region 1 Racks

Region 1 racks have poison panels on all peripheral walls facing other racks. Furthermore, the assembly distance across the gaps between Region 1 racks is larger than the assembly distance within the racks. Under abnormal conditions, in the event of lateral rack movement, the baseplate extensions will maintain a minimum rack to rack gap that is bounded by the infinite array calculations, and no further evaluations are necessary.

7.3.2 Gaps Between Region 2 Racks

Under normal conditions, the assembly distance across the gaps between Region 2 racks is larger than the assembly distance within these racks. Since there is at least one Boral panel between adjacent assemblies for these rack to rack interfaces, the condition in the gap is therefore bounded by the infinite array calculations, and no further evaluations are necessary.

7.3.3 Gaps Between Region 1 and Region 2 Racks

According to the data provided by Entergy, Region 1 and Region 2 are separated by distances that exceed the gaps between racks within either region, and therefore the condition is bounded by the infinite array calculations and no further evaluations are necessary.

7.3.4 Patterns Within Region 2 Racks.

The Region 2 racks are qualified for three types of fuel loading pattern: a uniform loading of spent fuel, a spent fuel checkerboard loading pattern, and a fresh fuel checkerboard loading pattern with empty cells. Within the Region 2 racks, various interfaces between these patterns are qualified. To show that the selected interfaces are acceptable, the following conditions are analyzed:

- An interface between the spent fuel uniform loading pattern and the spent fuel checkerboard. The configuration was chosen so that the high reactivity assembly in the spent fuel checkerboard pattern (5.0 wt%/27 GWD/MTU) is face adjacent to three low reactivity assemblies from the spent fuel checkerboard pattern (see Table 7.22), and face adjacent to 1 assembly meeting the uniform spent fuel requirement (see Table 7.22).
- Two interfaces are evaluated between checkerboards of spent fuel and fresh fuel/empty cells. The bounding case is the case where the fresh fuel assemblies face the high reactivity assembly in the spent fuel checkerboard pattern (5.0 wt%/27 GWD/MTU) on two sides, and has an empty cell on the other two sides. This condition bounds other interfaces between fresh and spent fuel, since the spent fuel with the highest permissible reactivity is used.

The interface configuration is acceptable, when the resulting k_{eff} is equivalent to, or less than the maximum k_{eff} of the individual pattern. The results are shown in Table 7.22 and show that this requirement is fulfilled for all analyzed cases and therefore:

- No restrictions are necessary between the uniform loading pattern and either of the checkerboard loading patterns (fresh or spent).
- For interfaces between the fresh fuel checkerboard and spent fuel checkerboard, the high reactivity spent fuel assembly (5.0 wt% ^{235}U, 27 GWD/MTU) may be face adjacent to no more than one fresh fuel assembly. The fresh fuel assembly may be face adjacent with up to 2 high reactivity spent fuel assemblies. Figure 7.5 shows one example of an acceptable 3x3 fresh fuel checkerboard within the center of a spent fuel checkerboard that meets these requirements.

7.4 Additional Calculations

7.4.1 Fuel Transfer Carriage Criticality

The transfer carriage is capable of accommodating two fuel assemblies at a time, carried in stainless steel boxes. The fuel transfer carriage is conservatively modeled as two fuel assemblies at 5.0 wt% ^{235}U and zero burnup separated by 5.06 inches of water only. The calculation of the criticality of the fuel transfer carriage accounts for both the carriage and the transfer tube. The results of the MCNP4a calculations are shown in Table 7.23.

Based on the design of the fuel transfer carriage, a fuel assembly could be mislocated outside the carriage. Two additional calculations were performed with a fresh fuel assembly mislocated directly adjacent to one of the two fuel assemblies in the carriage. The results of the MCNP4a calculations are shown in Table 7.23.

7.4.2 Upender Criticality

The criticality of the Upender is bounded by the calculation of the fuel transfer carriage in Section 7.4.1.

7.4.3 New Fuel Elevator Criticality

The criticality of the New Fuel Elevator is bounded by the calculation of the fuel transfer carriage in Section 7.4.1.

7.4.4 Boron Dilution Accident Evaluation

The soluble boron in the spent fuel pool water is conservatively assumed to contain a minimum of 1720 ppm under operating conditions. Significant loss or dilution of the soluble boron concentration is extremely unlikely, if not incredible. Nonetheless, an evaluation was performed based on the data provided by Entergy.

The required minimum soluble boron concentration is 447 ppm under normal conditions and 838 ppm for the most serious credible accident scenario (see Table 7.17 and Table 7.21). The volume of water in the pool is approximately 288,748 gallons. Large amounts of un-borated water would be necessary to reduce the boron concentration from 1720 ppm to 838 ppm or to 447 ppm. Abnormal or accident conditions are discussed below for either low dilution rates (abnormal conditions) or high dilution rates (accident conditions).

7.4.4.1 Low Flow Rate Dilution

Small dilution flow around pump seals and valve stems or mis-aligned valves could possibly occur in the normal soluble boron control system or related systems. Such failures might not be immediately detected. These flow rates would be of the order of 2 gpm maximum and the increased frequency of makeup flow might not be observed. However, an assumed loss flow-rate of 2 gpm dilution flow rate would require approximately 135 days to reduce the boron concentration to the minimum required 447 ppm under normal conditions or 72 days to reach the 838 ppm required for the most severe fuel handling accident. Routine surveillance measurements of the soluble boron concentration would readily detect the reduction in soluble boron concentration with ample time for corrective action.

Administrative controls require a measurement of the soluble boron concentration in the pool water at least weekly. Thus, the longest time period that a potential boron dilution might exist without a direct measurement of the boron concentration is 7 days. In this time period, an undetected dilution flow rate of 38.6 gpm would be required to reduce the boron concentration to 447 ppm. No known dilution flow rate of this magnitude has been identified. Further, a total of more than 389,000 gallons of un-borated water would be associated with the dilution event and such a large flow of un-borated water would be readily evident by high-level alarms and by visual inspection on daily walk-downs of the storage pool area.

7.4.4.2 High Flow Rate Dilution

Under certain accident conditions, it is conceivable that a high flow rate of un-borated water could flow into the spent fuel pool. As discussed in Section 5.6.4, the component cooling water makeup pumps provide makeup from the Condensate Storage Pool and have a capacity of 600 gpm. Such an accident scenario could result from the continuous operation of the Condensate Storage Pool pump and a flow rate of up to 600 gpm which could possibly contribute large amounts of un-borated water into the spent fuel.

Conservatively assuming that all the un-borated water from the pump poured into the pool and further assuming instantaneous mixing of the un-borated water with the pool water, it would take approximately 648 minutes to dilute the soluble boron concentration to 447 ppm, which is the minimum required concentration to maintain k_{eff} below 0.95 under normally operating conditions. In this dilution accident, some 389,000 gallons of water would be released into the spent fuel pool and multiple alarms would have alerted the control room of the accident consequences (including the fuel pool high-level alarm and the Fuel Handling Building sump high level alarm and Liquid Waste Management Trouble alarm). For this high flow rate condition, 346 minutes would be required to reach the 838 ppm required for the most severe fuel handling accident.

It is not considered credible that multiple alarms would fail or be ignored or that the spilling of large volumes of water would not be observed. Therefore, such a major failure would be detected in sufficient time for corrective action to avoid violation of an administrative guideline and to assure that the health and safety of the public is protected.

7.4.5 Temporary Storage Racks

The results of the TSR are summarized in Table 7.24. These results show that the TSR is qualified for loading fuel assemblies with an initial enrichment of up to 5.0 wt% ^{235}U. Based on information provided by Entergy, a fuel assembly may be mislocated on the exterior of the TSR. The mislocated fresh fuel assembly was modeled at the closest approach (See Table 5.5). For simplification, the following tolerances are included in the design basis model: fuel density, lattice pitch and enrichment (See Table 5.5). The results of the mislocated case and the necessary soluble boron amount are present in Table 7.24.

7.4.6 Fuel Pin Storage Container

The FPSC calculation involved comparing the reactivity of the FPSC to three cases of NGF fuel assemblies under equivalent modeling conditions: a fresh fuel assembly, a burnup of 27 GWD/MTU and a burnup of 33.4 GWD/MTU, all at 5.0 wt% ^{235}U. These three cases match the most reactive fuel assembly for the three loading patterns analyzed in the main body of the report. The results of these comparisons can be seen in Table 7.25. Therefore the FPSC can be placed in any location intended for fresh or spent fuel.

7.4.7 New Fuel Storage Vault

The maximum calculated reactivity of the NFV is listed in Table 7.28. The calculated reactivity as a function of water density is also shown in Figure 7.7. The results show that the optimum moderator density occurs at 100% water density and this maximum k_{eff} is below the regulatory limit.

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3. "Lumped Fission Product and Pm148m Cross Sections for MCNP," Holtec Report HI-2033031, Rev 0, September 2003.

4. M. Edenius, K. Ekberg, B.H. Forssén, and D. Knott, "CASMO-4 A Fuel Assembly Burnup Program User's Manual," Studsvik/SOA-95/1, Studsvik of America, Inc. and Studsvik Core Analysis AB (proprietary).

5. D. Knott, "CASMO-4 Benchmark Against Critical Experiments," SOA-94/13, Studsvik of America, Inc., (proprietary).

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

8. ANS-8.1/N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," April 14, 1975.

9. S.E. Turner, "Uncertainty Analysis - Burnup Distributions," presented at the DOE/SANDIA Technical Meeting on Fuel Burnup Credit, Special Session, ANS/ENS Conference, Washington, D.C., November 2, 1988.

10. "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit," NUREG/CR-6760, ORNL/TM-2000-321, March 2002.

[1] Note: The revision status of Holtec documents cited above is subject to updates as the project progresses. This document will be revised if a revision to any of the above-referenced Holtec work materially affects the instructions, results, conclusions or analyses contained in this document. Otherwise, a revision to this document will not be made and the latest revision of the referenced Holtec documents shall be assumed to supercede the revision numbers cited above. The Holtec Project Manager bears the undivided responsibility to ensure that there is no intra-document conflict with respect to the information contained in all Holtec generated documents on a safety-significant project. The latest revision number of all documents produced by Holtec International in a safety significant project is readily available from the company's electronic network.

Table 5.1
Fuel Assembly Specification

Assembly Type	16x16 Standard	16x16 NGF
Stack Density, g/cm^3	10.412	10.522
Fuel Rod Pitch, in	0.506	0.506
Number of Fuel Rods	236	236
Number of Guide Tubes	5	5
Fuel Rod Clad OD, in	0.382	0.374
Fuel Rod Clad ID, in	0.332	0.329
Active Length, in	149.61-150.0	150.0
Fuel Pellet Diameter, in	0.325	0.3225
Guide Tube OD, in	0.98	0.98
Guide Tube ID, in	0.9	0.9
ZrB_2 Rod Coating Loading (mgm ^{10}B/inch)	3.14	3.14
ZrB_2 Rod Coating Thickness (inches)	0.0004167	0.000417
ZrB_2 Rod Coating Length (inches)	136	138
Fuel Assembly Width (min), in.	n/a	8.125
Bottom of Active Fuel to Bottom of Fuel Assembly, in.	n/a	5.402

Table 5.2
Core Operating Parameter for Depletion Analyses

Parameter	Value
Soluble Boron Concentration (bounding cycle average), ppm	1000
Reactor Specific Power, MW/MTU	40.5
Core Average Fuel Temperature, °F	1041.0
Core Average Moderator Temperature at the Top of the Active Region, °F	614.0
In-Core Assembly Pitch, Inches	8.18

Table 5.3
Axial Burnup Profiles

Node Center (cm)	Relative Burnup ≤ 25 GWD/MT	Relative Burnup > 25 GWD/MT
7.62	0.54	0.593
22.86	0.773	0.819
38.1	0.921	0.961
53.34	1.013	1.028
68.58	1.055	1.051
83.82	1.065	1.057
99.06	1.064	1.058
114.3	1.061	1.058
129.54	1.058	1.057
144.78	1.056	1.056
160.02	1.054	1.055
175.26	1.053	1.054
190.5	1.052	1.054
205.74	1.051	1.053
220.98	1.05	1.051
236.22	1.047	1.049
251.46	1.046	1.048
266.7	1.044	1.046
281.94	1.04	1.043
297.18	1.031	1.036
312.42	0.994	1.021
327.66	0.92	0.966
342.9	0.81	0.873
358.14	0.655	0.725
373.38	0.441	0.508

Table 5.4

Storage Rack and Spent Fuel Pool Parameter Specification

Region 1		
Parameter	**Value**	**Tolerance**
Cell ID, in	8.5	
Cell Wall thickness, in	0.075	
[2]Cell Pitch, in	10.185	
Boundary Sheathing Thickness, in	0.075	
Inner Sheathing Thickness, in	0.0235	
[3]Poison Thickness, in	0.089	
Poison Width, in	7.25	
Poison Gap, (nominal) in	0.096	
Flux Trap (nominal) in	1.3	
B-10 Loading, (nom) g/cm^2	0.028	
Region 2		
Parameter	**Value**	**Tolerance**
Cell ID, in	8.5	
Cell Wall thickness, in	0.075	
Cell Pitch, in	8.692	
Boundary Sheathing Thickness, in	0.075	
Inner Sheathing Thickness, in	0.035	
Poison Thickness, in	0.075	
Poison Width, in	7.25	
Poison Gap, in (nominal)	0.082	
B-10 Loading, (nom) g/cm^2	0.0216	
Additional Spent Fuel Pool Information		
Parameter	**Value**	**Tolerance**
Soluble Boron Concentration, ppm	1720	n/a
Spent Fuel Pool Volume, cf	38,600	n/a
Fuel Transfer Carriage Gap, in	5.06	n/a
Refueling Water Storage Pool (min), ppm	2050	n/a
Refueling Water Pool Purification Pump, gpm	150	n/a
Component Cooling Water Makeup Pumps, gpm	600	n/a

[2] Note that [4] indicates a larger cell-cell pitch for the North-South direction. The value used is bounding.

[3] Note that the actual model used 0.075 inches for the poison thickness for conservatism.

Table 5.5

Reactor Building Temporary Storage Rack

Parameter	Value
Number of Storage Cells	5
Pitch, in.	18 ± 0.02
Rack Opening, in.	8.62 ±0.06
Canal Wall to Cell Center, in.	8.06
Distance from Outside Edge of Cell Wall to Outside Edge of Structural Material of Cell, in.	2.25
Enrichment Tolerance, wt% ^{235}U	± 0.05
Fuel Density Tolerance, g UO_2/cm^3	±0.165
Rack Pitch Spacing[4] Tolerance, in.	±0.555

Fuel Pin Storage Container

Parameter	Value
Steel Tube Outer Diameter[5], in.	0.625
Steel Tube Thickness, in.	0.035
Steel Tube Pitch, in.	0.917

[4] The rack pitch spacing is used to account for the possible gaps between the fuel assembly and rack inner wall. This value is used in the place of the much smaller pitch tolerance listed.

[5] Note: 4 tubes have a larger outer diameter; the smaller diameter is used to conservatively model less steel.

Table 5.6

New Fuel Vault Parameters

Parameter	Value
Vault North-South width, ft.	27.5
Vault East-West width, ft.	29.25
Rack Cell Opening, in.	8.9375
Thickness of Redwood Planks, in.	1.5
Rack Cell Pitch, in.	21
East-West Rack Module Center-to-Center Cell Separation, in.	49
North-South Rack Module Center-to-Center Cell Separation, in.	58
Distance from Fuel Assembly Center to North Wall, in.	12.25
Distance from Fuel Assembly Center to East and West Wall, in.	60
Distance from Fuel Assembly Center to South Wall, in.	91.75
Depth of Rack Cell, in.	190

Table 7.1

Results of the Region 1 Reference Fuel Assembly Calculations

Assembly Type at 5.0 wt% ^{235}U	Calculated k_{eff}
Standard	0.9164
NGF	0.9268

Table 7.2

Region 1 Manufacturing Tolerances and Uncertainty Calculations

Parameter	Calculated k_{eff}	Delta-k
Reference Case CASMO	0.9268	n/a
Storage Cell ID Increase	0.9370	0.0102
Storage Cell ID Decrease	0.9205	-0.0063
Storage Cell Pitch Increase	0.9184	-0.0084
Storage Cell Pitch Decrease	0.9350	0.0082
Storage Cell Poison Width Increase	0.9250	-0.0018
Storage Cell Poison Width Decrease	0.9289	0.0021
Storage Cell Poison Gap Minimum	0.9263	-0.0005
Storage Cell Box Wall Decrease	0.9242	-0.0026
Storage Cell Box Wall Increase	0.9285	0.0017
Storage Cell Poison B-10 Loading Minimum	0.9291	0.0023
Fuel Rod Pitch Increase	0.9277	0.0009
Fuel Rod Pitch Decrease	0.9259	-0.0009
Fuel Rod Clad OD Increase	0.9248	-0.0020
Fuel Rod Clad OD Decrease	0.9288	0.0020
Fuel Rod Clad Thickness Minimum	0.9267	-0.0001
Fuel Pellet OD Increase	0.9271	0.0003
Fuel Pellet OD Decrease	0.9265	-0.0003
Guide Tube OD Increase	0.9268	0.0000
Guide Tube OD Decrease	0.9268	0.0000
Guide Tube Thickness Minimum	0.9272	0.0004
Fuel Pellet Enrichment Increase	0.9284	0.0016
Fuel Pellet Density Increase	0.9285	0.0017
Statistical Combination of Positive Reactivity Uncertainties:		**0.0140**

Table 7.3

Region 1 Temperature and Water Density Effects Results

Case	Calculated k_{eff}	Delta-k
Reference Temperature 32 F	0.9268	0.0000
39.2 F	0.9266	-0.0002
68 F	0.9253	-0.0015
80.33 F	0.9244	-0.0024
140 F	0.9188	-0.0080
255 F 0% voids	0.9028	-0.0240
255 F 10% voids	0.8681	-0.0587
255 F 20% voids	0.8295	-0.0973
Bias to 80.33 F:		**0.0024**

Table 7.4

Summary of the Criticality Safety Analysis for Region 1 Without Soluble Boron

Uncertainties:	
Bias Uncertainty (95%/95%)	± 0.0011
Calculation Statistics (95%/95%,2.0xσ)	± 0.0014
Fuel Eccentricity	Negative
Manufacturing Tolerances	± 0.0140
Statistical Combination of Uncertainties	± 0.0141
Reference Calculated k_{eff} (MCNP4a)	0.9354
Total Uncertainty (above)	0.0141
Bias to 80.33 °F	0.0024
Calculation Bias (see Appendix A)	0.0009
Maximum k_{eff}	**0.9527**
Regulatory Limit k_{eff}	1.0000

Table 7.5

Summary of the Criticality Safety Analysis for Region 1 with Soluble Boron Requirement

Soluble Boron ppm	60.7
Uncertainties:	
Bias Uncertainty (95%/95%)	± 0.0011
Calculation Statistics (95%/95%,2.0xσ)	± 0.0014
Fuel Eccentricity	Negative
Manufacturing Tolerances	± 0.0140
Statistical Combination of Uncertainties	± 0.0141
Reference Calculated k_{eff} (MCNP4a)	0.9277
Total Uncertainty (above)	0.0141
Bias to 80.33 °F	0.0024
Calculation Bias (see Appendix A)	0.0009
Maximum k_{eff}	**0.9450**
Regulatory Limit k_{eff}	0.9500

Table 7.6

Results of Associated Region 1 Reactivity Calculations

Eccentric Positioning Case	
Case	**k_{eff}**
Reference	0.9354
Eccentric	0.9332
Delta-k	-0.0022
Soluble Boron Case	
ppm Boron	**k_{eff}**
0	0.9354
200	0.9099
Target k_{eff}	0.9277
Calculated ppm	61
Mislocated FA Case	
ppm Boron	**k_{eff}**
0	0.9510
400	0.8962
Target k_{eff}	0.9277
Calculated ppm	170

Table 7.7
Region 2 Calculations for the Reference Fuel Assembly

Enrichment		2.0 wt% ^{235}U	
Burnup (GWD/MTU)	**Standard**	**NGF**	**Δk**
0.0	0.9568	0.9631	0.0063
0.1	0.9537	0.9600	0.0063
2.0	0.9391	0.9448	0.0057
4.0	0.9231	0.9283	0.0052

Enrichment		3.5 wt% ^{235}U	
Burnup (GWD/MTU)	**Standard**	**NGF**	**Δk**
0.0	1.1113	1.1179	0.0067
0.1	1.1089	1.1156	0.0067
2.0	1.0887	1.0952	0.0064
4.0	1.0719	1.0782	0.0062
6.0	1.0547	1.0607	0.0061
8.0	1.0377	1.0435	0.0058
10.0	1.0211	1.0267	0.0055
11.0	1.0130	1.0184	0.0054
12.5	1.0012	1.0063	0.0052
15.0	0.9819	0.9867	0.0048
17.5	0.9631	0.9674	0.0043
20.0	0.9446	0.9484	0.0038
22.5	0.9265	0.9298	0.0033
25.0	0.9088	0.9115	0.0027

Table 7.7 Continued

Burnup (GWD/MTU)	Standard	NGF	Δk
Enrichment		5.0 wt% ^{235}U	
0.0	1.1932	1.1998	0.0066
0.1	1.1914	1.1980	0.0066
2.0	1.1708	1.1773	0.0065
4.0	1.1558	1.1623	0.0064
6.0	1.1406	1.1470	0.0064
8.0	1.1254	1.1317	0.0063
10.0	1.1106	1.1168	0.0062
11.0	1.1034	1.1095	0.0062
12.5	1.0927	1.0987	0.0060
15.0	1.0753	1.0812	0.0059
17.5	1.0584	1.0640	0.0056
20.0	1.0417	1.0471	0.0054
22.5	1.0254	1.0305	0.0051
25.0	1.0093	1.0141	0.0048
27.5	0.9934	0.9979	0.0044
30.0	0.9776	0.9817	0.0041
32.5	0.9620	0.9656	0.0037
35.0	0.9464	0.9497	0.0032
37.5	0.9310	0.9338	0.0028
40.0	0.9157	0.9180	0.0023
42.5	0.9005	0.9023	0.0018

Table 7.8

Region 2 Calculations for NGF Fuel IFBA Rods Reactivity Effect

wt% U235	3.5			5.0		
Number of IFBA Rods / Burnup GWD/MTU	0	148	Delta k	0	148	Delta k
0.0	1.1179	0.8007	-0.3172	1.1998	0.9152	-0.2846
0.1	1.1156	0.8026	-0.3130	1.1980	0.9162	-0.2818
2.0	1.0952	0.8564	-0.2388	1.1773	0.9476	-0.2297
4.0	1.0782	0.9013	-0.1769	1.1623	0.9774	-0.1848
6.0	1.0607	0.9330	-0.1278	1.1470	1.0000	-0.1469
8.0	1.0435	0.9537	-0.0898	1.1317	1.0165	-0.1153
10.0	1.0267	0.9655	-0.0611	1.1168	1.0276	-0.0892
11.0	1.0184	0.9686	-0.0498	1.1095	1.0315	-0.0780
12.5	1.0063	0.9704	-0.0359	1.0987	1.0353	-0.0635
15.0	0.9867	0.9673	-0.0194	1.0812	1.0371	-0.0441
17.5	0.9674	0.9585	-0.0089	1.0640	1.0343	-0.0297
20.0	0.9484	0.9461	-0.0024	1.0471	1.0279	-0.0192
22.5	0.9298	0.9315	0.0017	1.0305	1.0188	-0.0117
25.0	0.9115	0.9156	0.0041	1.0141	1.0076	-0.0065
27.5	0.8935	0.8990	0.0055	0.9979	0.9951	-0.0028
30.0	0.8758	0.8821	0.0063	0.9817	0.9815	-0.0002
32.5	0.8585	0.8653	0.0067	0.9656	0.9673	0.0016
35.0	0.8417	0.8486	0.0069	0.9497	0.9525	0.0029
37.5	0.8253	0.8323	0.0070	0.9338	0.9375	0.0037
40.0	0.8095	0.8165	0.0070	0.9180	0.9223	0.0043
42.5	0.7942	0.8011	0.0069	0.9023	0.9070	0.0047
45.0	0.7796	0.7864	0.0068	0.8868	0.8918	0.0050
47.5	n/a			0.8714	0.8766	0.0052
50.0	n/a			0.8562	0.8616	0.0053
52.5	n/a			0.8413	0.8468	0.0054
55.0	n/a			0.8267	0.8322	0.0055
57.5	n/a			0.8125	0.8180	0.0055
60.0	n/a			0.7986	0.8041	0.0055

Table 7.9

Region 2 Calculations for Eccentric Fuel Positioning

Case	Calculated k_{eff}	Delta k
Reference Uniform Loading	0.9570	-0.0053
Spent Fuel Uniform Loading Eccentric Positioning	0.9517	
Reference Spent Fuel Checkerboard Loading	0.9719	-0.0044
Spent Fuel Checkerboard Loading Eccentric Positioning	0.9675	
Reference Fresh Checkerboard	0.8256	-0.0032
Fresh Fuel Checkerboard Eccentric Positioning	0.8224	

Table 7.10

Region 2 Calculations for Manufacturing Tolerance Uncertainties for Fuel Storage Cell

Burnup GWD/MTU	Enrichment	Ref Case	ID +	ID -	Poison Width +	Poison Width -	Poison Gap Min	Box Wall +	Box Wall -	B-10 Loading Min	Statistical Combo
0.0	2	0.9631	-0.0023	-0.0013	-0.0020	0.0026	0.0001	0.0001	-0.0001	0.0034	0.0045
2.0	2	0.9448	-0.0024	-0.0012	-0.0020	0.0025	0.0001	0.0001	-0.0001	0.0034	0.0043
4.0	2.5	0.9897	-0.0029	-0.0009	-0.0021	0.0026	0.0001	0.0001	-0.0001	0.0035	0.0045
8.0	2.5	0.9534	-0.0028	-0.0008	-0.0020	0.0025	0.0001	0.0001	-0.0001	0.0034	0.0043
11.0	3	0.9769	-0.0030	-0.0006	-0.0021	0.0025	0.0001	0.0000	-0.0001	0.0035	0.0043
15.0	3	0.9443	-0.0029	-0.0006	-0.0020	0.0024	0.0001	0.0001	-0.0001	0.0034	0.0042
15.0	3.5	0.9867	-0.0032	-0.0004	-0.0021	0.0026	0.0001	0.0001	-0.0001	0.0035	0.0044
22.5	3.5	0.9298	-0.0029	-0.0005	-0.0020	0.0024	0.0001	0.0000	-0.0001	0.0033	0.0041
22.5	4	0.9679	-0.0032	-0.0003	-0.0020	0.0025	0.0001	0.0000	-0.0001	0.0034	0.0043
27.5	4	0.9326	-0.0030	-0.0004	-0.0020	0.0024	0.0001	0.0000	-0.0001	0.0033	0.0041
27.5	4.5	0.9673	-0.0032	-0.0002	-0.0020	0.0025	0.0001	0.0000	-0.0001	0.0034	0.0042
32.5	4.5	0.9338	-0.0031	-0.0003	-0.0020	0.0024	0.0001	0.0000	-0.0001	0.0033	0.0041
32.5	5	0.9656	-0.0033	-0.0002	-0.0020	0.0025	0.0001	0.0000	-0.0001	0.0034	0.0042
40.0	5	0.9180	-0.0030	-0.0002	-0.0019	0.0024	0.0001	0.0000	-0.0001	0.0032	0.0040

Table 7.11
Region 2 Calculations for Fuel Tolerance Uncertainties

Burnup GWD/ MTU	Enr	Ref Case	Pitch +	Pitch -	Clad OD +	Clad OD -	Clad Thickness Min	Fuel Pellet OD +	Fuel Pellet OD -	Guide Tube OD +	Guide Tube OD -	Guide Tube Thickness Min	Fuel Pellet Enr +	Fuel Pellet Density +	Statistical Combo.
0.0	2.0	0.9631	0.0007	-0.0007	-0.0009	0.0009	0.0005	0.0004	-0.0004	0.0000	0.0000	0.0002	0.0074	0.0022	0.0079
2.0	2.0	0.9448	0.0007	-0.0007	-0.0008	0.0008	0.0005	0.0004	-0.0004	0.0000	0.0000	0.0002	0.0070	0.0022	0.0075
4.0	2.5	0.9897	0.0008	-0.0008	-0.0008	0.0008	0.0005	0.0004	-0.0004	0.0000	0.0000	0.0002	0.0054	0.0019	0.0059
8.0	2.5	0.9534	0.0008	-0.0008	-0.0007	0.0007	0.0005	0.0004	-0.0004	0.0000	0.0000	0.0002	0.0054	0.0020	0.0059
11.0	3.0	0.9769	0.0009	-0.0009	-0.0008	0.0007	0.0005	0.0003	-0.0004	0.0000	0.0000	0.0002	0.0045	0.0018	0.0050
15.0	3.0	0.9443	0.0008	-0.0008	-0.0006	0.0006	0.0004	0.0004	-0.0004	0.0000	0.0000	0.0002	0.0046	0.0020	0.0051
15.0	3.5	0.9867	0.0009	-0.0009	-0.0007	0.0007	0.0004	0.0003	-0.0003	0.0000	0.0000	0.0002	0.0039	0.0017	0.0044
22.5	3.5	0.9298	0.0009	-0.0008	-0.0005	0.0005	0.0004	0.0004	-0.0004	0.0000	0.0000	0.0001	0.0041	0.0020	0.0047
22.5	4.0	0.9679	0.0009	-0.0009	-0.0007	0.0006	0.0004	0.0003	-0.0004	0.0000	0.0000	0.0002	0.0035	0.0017	0.0041
27.5	4.0	0.9326	0.0009	-0.0009	-0.0005	0.0005	0.0004	0.0004	-0.0004	0.0000	0.0000	0.0001	0.0037	0.0019	0.0043
27.5	4.5	0.9673	0.0010	-0.0009	-0.0006	0.0006	0.0004	0.0003	-0.0003	0.0000	0.0000	0.0002	0.0032	0.0016	0.0038
32.5	4.5	0.9338	0.0009	-0.0009	-0.0005	0.0005	0.0004	0.0004	-0.0004	0.0000	0.0000	0.0001	0.0033	0.0018	0.0040
32.5	5.0	0.9656	0.0010	-0.0010	-0.0006	0.0006	0.0004	0.0003	-0.0003	0.0000	0.0000	0.0002	0.0030	0.0015	0.0036
40.0	5.0	0.9180	0.0009	-0.0009	-0.0004	0.0004	0.0004	0.0004	-0.0004	0.0000	0.0000	0.0001	0.0031	0.0019	0.0039

Table 7.12

Region 2 Calculations for Pool Temperature Tolerance Uncertainties

Burnup GWD/MTU	Enr	Ref Case T = 32 F	T = 39.2 F	T = 80.33 F	T = 255 F, 0% Voids	T = 255 F, 10% Voids	T = 255 F, 20% Voids
0.0	2.0	0.9631	-0.0008	-0.0056	-0.0318	-0.0495	-0.0714
2.0	2.0	0.9448	-0.0007	-0.0051	-0.0291	-0.0462	-0.0675
4.0	2.5	0.9897	-0.0006	-0.0046	-0.0273	-0.0458	-0.0684
8.0	2.5	0.9534	-0.0005	-0.0041	-0.0248	-0.0431	-0.0655
11.0	3.0	0.9769	-0.0005	-0.0038	-0.0242	-0.0435	-0.0667
15.0	3.0	0.9443	-0.0004	-0.0035	-0.0225	-0.0414	-0.0643
15.0	3.5	0.9867	-0.0004	-0.0036	-0.0234	-0.0433	-0.0671
22.5	3.5	0.9298	-0.0004	-0.0031	-0.0208	-0.0400	-0.0631
22.5	4.0	0.9679	-0.0004	-0.0032	-0.0219	-0.0419	-0.0658
27.5	4.0	0.9326	-0.0003	-0.0029	-0.0203	-0.0399	-0.0633
27.5	4.5	0.9673	-0.0003	-0.0030	-0.0213	-0.0416	-0.0658
32.5	4.5	0.9338	-0.0003	-0.0028	-0.0199	-0.0398	-0.0635
32.5	5.0	0.9656	-0.0003	-0.0029	-0.0208	-0.0414	-0.0657
40.0	5.0	0.9180	-0.0003	-0.0025	-0.0189	-0.0388	-0.0625

Table 7.13

Region 2 Results for the Spent Fuel Uniform Loading

Enrichment (wt% ^{235}U)	2.0	2.5	3.0	3.5	4.0	4.5	5.0
Burnup (GWD/MTU)	0	5.89	11.77	17.50	23.55	28.15	33.39
CASMO Burnup for Tolerances	0.0	4.0	11.0	15.0	22.5	27.5	32.5
Depletion Uncertainty	0.0000	0.0019	0.0051	0.0066	0.0091	0.0105	0.0117
Manufacturing Uncertainty	0.0045	0.0045	0.0043	0.0044	0.0043	0.0042	0.0042
Fuel Uncertainty	0.0079	0.0059	0.0050	0.0044	0.0041	0.0038	0.0036
Calculational Uncertainty	0.0012	0.0012	0.0014	0.0014	0.0012	0.0012	0.0014
Code Uncertainty	0.0011	0.0011	0.0011	0.0011	0.0011	0.0011	0.0011
Total Uncertainty	0.0092	0.0078	0.0085	0.0092	0.0110	0.0121	0.0131
Code Bias	0.0009	0.0009	0.0009	0.0009	0.0009	0.0009	0.0009
Temperature Bias	0.0056	0.0046	0.0038	0.0036	0.0032	0.0030	0.0029
IFBA Bias	0.0070	0.0070	0.0070	0.0070	0.0070	0.0070	0.0070
MCNP k_{eff} 0 ppm Boron	0.9613	0.9747	0.9747	0.9743	0.9729	0.9720	0.9712
MCNP k_{eff} 600 ppm Boron	0.8560	n/a	n/a	0.8948	n/a	n/a	0.9040
Max k_{eff} 0 ppm Boron	0.9950	0.9950	0.9950	0.9950	0.9950	0.9950	0.9950
Max k_{eff} with 600 ppm Boron	0.8787	n/a	n/a	0.9175	n/a	n/a	0.9267

Table 7.14

Region 2 Results for the Spent Fuel Checkerboard Loading

Enrichment (wt% ^{235}U)	2.0	2.5	3.0	3.5	4.0	4.5	5.0
Burnup (GWD/MTU)	2.41	9.42	16.21	23.55	29.53	34.64	41.23
CASMO Burnup for Tolerances	2.0	8.0	15.0	22.5	27.5	32.5	40.0
Depletion Uncertainty	0.0009	0.0038	0.0067	0.0094	0.0109	0.0122	0.0141
Manufacturing Uncertainty	0.0043	0.0043	0.0042	0.0041	0.0041	0.0041	0.0040
Fuel Uncertainty	0.0075	0.0059	0.0051	0.0047	0.0043	0.0040	0.0039
Calculational Uncertainty	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012	0.0012
Code Uncertainty	0.0011	0.0011	0.0011	0.0011	0.0011	0.0011	0.0011
Total Uncertainty	0.0088	0.0083	0.0096	0.0114	0.0125	0.0135	0.0152
Code Bias	0.0009	0.0009	0.0009	0.0009	0.0009	0.0009	0.0009
Temperature Bias	0.0051	0.0041	0.0035	0.0031	0.0029	0.0028	0.0025
IFBA Bias	0.0070	0.0070	0.0070	0.0070	0.0070	0.0070	0.0070
MCNP k_{eff} 0 ppm Boron	0.9731	0.9747	0.9740	0.9726	0.9717	0.9708	0.9693
MCNP k_{eff} 600 ppm Boron	0.8893	n/a	n/a	0.8969	n/a	n/a	0.9008
Max k_{eff} 0 ppm Boron	0.9950	0.9950	0.9950	0.9950	0.9950	0.9950	0.9950
Max k_{eff} with 600 ppm Boron	0.9112	n/a	n/a	0.9193	n/a	n/a	0.9265

Table 7.15
Region 2 Results for the Fresh Checkerboard Loading,
5.0 wt% ^{235}U

Enrichment (wt% ^{235}U)	5.0
Burnup (GWD/MTU)	0
Manufacturing Uncertainty	0.0053
Fuel Uncertainty	0.0029
Calculational Uncertainty	0.0014
Code Uncertainty	0.0011
Total Uncertainty	0.0063
Code Bias	0.0009
Temperature Bias	0.0034
IFBA Bias	0.0070
MCNP k_{eff} 0 ppm Boron	0.8256
Max k_{eff} without Boron	0.8432

Table 7.16

Summary of the Criticality Safety Analysis for Region 2, Spent Fuel Uniform Loading, 0 ppm Soluble Boron

Enrichment (wt% ^{235}U)	5.0		
Burnup (GWD/MTU)	33.4		
Soluble Boron ppm	0.0		
Fuel Eccentricity	negative		
Statistical Combination of Uncertainties		±	0.0131
Calculated k_{eff} (MCNP4a)			0.9712
IFBA Bias			0.0070
Bias to 80.33 °F			0.0029
Calculation Bias (see Appendix A)			0.0009
Maximum k_{eff}			**0.9950**
Regulatory Limit k_{eff}			1.0000

Table 7.17

Summary of the Criticality Safety Analysis for Region 2, Spent Fuel Uniform Loading, 447 ppm Soluble Boron

Enrichment (wt% ^{235}U)	5.0		
Burnup (GWD/MTU)	33.4		
Soluble Boron (ppm)	447		
Statistical Combination of Uncertainties		±	0.0131
Calculated k_{eff} (MCNP4a)			0.9212
IFBA Bias			0.0070
Bias to 80.33 °F			0.0029
Calculation Bias (see Appendix A)			0.0009
Maximum k_{eff}			**0.9450**
Regulatory Limit k_{eff}			0.9500

Table 7.18

Summary of the Criticality Safety Analysis for Region 2, Spent Fuel Checkerboard Loading, 0 ppm Soluble Boron

Enrichment (wt% ^{235}U)	5.0	
Burnup (GWD/MTU)	41.2	
Soluble Boron (ppm)	0.0	
Fuel Eccentricity	negative	
Statistical Combination of Uncertainties	±	0.0152
Calculated k_{eff} (MCNP4a)		0.9693
IFBA Bias		0.0070
Bias to 80.33 °F		0.0025
Calculation Bias (see Appendix A)		0.0009
Maximum k_{eff}		**0.9950**
Regulatory Limit k_{eff}		1.0000

Table 7.19

Summary of the Criticality Safety Analysis for Region 2, Spent Fuel Checkerboard Loading, 438 ppm Soluble Boron

Enrichment (wt% ^{235}U)	5.0	
Burnup (GWD/MTU)	41.2	
Soluble Boron (ppm)	438	
Statistical Combination of Uncertainties	±	0.0152
Calculated k_{eff} (MCNP4a)		0.9193
IFBA Bias		0.0070
Bias to 80.33 °F		0.0025
Calculation Bias (see Appendix A)		0.0009
Maximum k_{eff}		**0.9450**
Regulatory Limit k_{eff}		0.9500

Table 7.20

Summary of the Criticality Safety Analysis for Region 2, Fresh Fuel Checkerboard Loading , 0 ppm Soluble Boron

Enrichment (wt% ^{235}U)	5.0	
Burnup (GWD/MTU)	0.0	
Soluble Boron (ppm)	0.0	
Fuel Eccentricity	negative	
Statistical Combination of Uncertainties		± 0.0063
Calculated k_{eff} (MCNP4a)		0.8256
IFBA Bias		0.0070
Bias to 80.33 °F		0.0034
Calculation Bias (see Appendix A)		0.0009
Maximum k_{eff}		**0.8432**
Regulatory Limit k_{eff}		1.0000

Table 7.21

Summary of Region 2 Accident Cases

Case	Result
Dropped Fuel Assembly - Horizontal On Top of Cells	Negligible
Dropped Fuel Assembly - Vertical into Storage Cell	Negligible
Misloaded Fuel Assembly, Spent Fuel Checkerboard Loading, 5.0 wt% ^{235}U (ppm Soluble Boron)	838[6]
Mislocated Fuel Assembly, Spent Fuel Checkerboard Loading, 5.0 wt% ^{235}U (ppm Soluble Boron)	534[7]

[6] This case was the maximum for the misloaded assembly in the spent fuel uniform loading, spent fuel checkerboard loading, or fresh fuel checkerboard.
[7] This case was the maximum for the mislocated assembly in the spent fuel uniform loading, spent fuel checkerboard loading, or fresh fuel checkerboard.

Table 7.22

Region 2 Calculation Results for the Interface Cases

Description		Axial Profile	Enr	Burnup (GWD/MTU)	k_{eff}	Ref k_{eff} (at curve)
Interface between half a rack of fresh fuel checkerboard and half a rack of spent fuel checkerboard	Spent fuel checkerboard loading, fresh FA adjacent 27 GWD/MTU, 5.0 wt% ^{235}U FA	Segmented	2.0	2.41	0.9598	0.9731
		Segmented	3.5	23.55	0.9513	0.9726
		Segmented	5.0	41.23	0.9464	0.9693
		Uniform	2.0	2.41	0.959	0.9731
		Uniform	3.5	23.55	0.9539	0.9726
		Uniform	5.0	41.23	0.9506	0.9693
Interface between a 3x3 set of fresh checkerboard (fresh in center) surrounded by a rack of spent fuel checkerboard	Spent fuel checkerboard loading, fresh FA adjacent 27 GWD/MTU, 5.0 wt% ^{235}U FA	Segmented	2.0	2.41	0.9734	0.9731
		Segmented	3.5	23.55	0.9659	0.9726
		Segmented	5.0	41.23	0.9611	0.9693
		Uniform	2.0	2.41	0.9716	0.9731
		Uniform	3.5	23.55	0.9679	0.9726
		Uniform	5.0	41.23	0.9642	0.9693
Interface between a set of spent fuel checkerboard loading fuel and spent uniform loading fuel.		Segmented	2.0	2.41	0.9697	0.9731
		Segmented	3.5	23.55	0.9694	0.9726
		Segmented	5.0	41.23	0.9667	0.9693
		Uniform	2.0	2.41	0.97	0.9731
		Uniform	3.5	23.55	0.9706	0.9726
		Uniform	5.0	41.23	0.9653	0.9693

Table 7.23

Results of the Calculation of the Fuel Transfer Carriage

Description	Calculated k_{eff}
Reference Case	0.9436
Mislocated Case	1.0612
800 ppm Boron Case	0.9209

Table 7.24

Results of the Criticality Analysis for the TSR

Description	Calculated k_{eff}
TSR Design Basis Model	0.9297
TSR Mislocated Fuel Assembly Model	1.0204
TSR Mislocated Fuel Assembly Model with 800 ppm Soluble Boron	0.8525
Extrapolated TSR Soluble Boron Requirement for Mislocated Accident, ppm	359

Table 7.25

Results of the Criticality Analysis for the FPSC

Description	Calculated k_{eff}
FPSC Design Basis Model	0.6715
5.0 wt% ^{235}U Fuel Assembly at 33.4 GWD/MTU	0.7521
5.0 wt% ^{235}U Fuel Assembly at 27 GWD/MTU	0.7784
Fresh NGF Fuel Assembly	0.9226

Table 7.26

Region 2 Burnup Versus Enrichment Curve for Spent Fuel Uniform Loading

Enrichment (wt% ^{235}U)	**Burnup (GWD/MTU)**
2.0	0.0
2.5	5.9
3.0	11.8
3.5	17.5
4.0	23.5
4.5	28.1
5.0	33.4

Table 7.27

Region 2 Burnup Versus Enrichment Curve for Spent Fuel Checkerboard Loading

Enrichment (wt% ^{235}U)	Burnup (GWD/MTU)
2.0	2.4
2.5	9.4
3.0	16.2
3.5	23.6
4.0	29.5
4.5	34.6
5.0	41.2

Table 7.28

Summary of the Criticality Safety Analysis for New Fuel Vault, 100% Moderator Density

Tolerances:					
Enrichment k_{eff}	0.9195	±	0.0008		
Enrichment Uncertainty				±	0.0034
Pellet Density k_{eff}	0.9192	±	0.0008		
Pellet Density Uncertainty				±	0.0031
Storage Rack Pitch k_{eff}	0.9187	±	0.0007		
Storage Rack Pitch Uncertainty				±	0.0023
Bias Uncertainty (95%/95%)				±	0.0011
Calculation Statistics (95%/95%,2xσ)				±	0.0014
Statistical Combination of Uncertainties				±	0.0054
Calculated k_{eff} (MCNP4a)					0.9184
Calculation Bias (see Appendix A)					0.0009
Maximum k_{eff}					**0.9247**
Regulatory Limit k_{eff}					0.9500

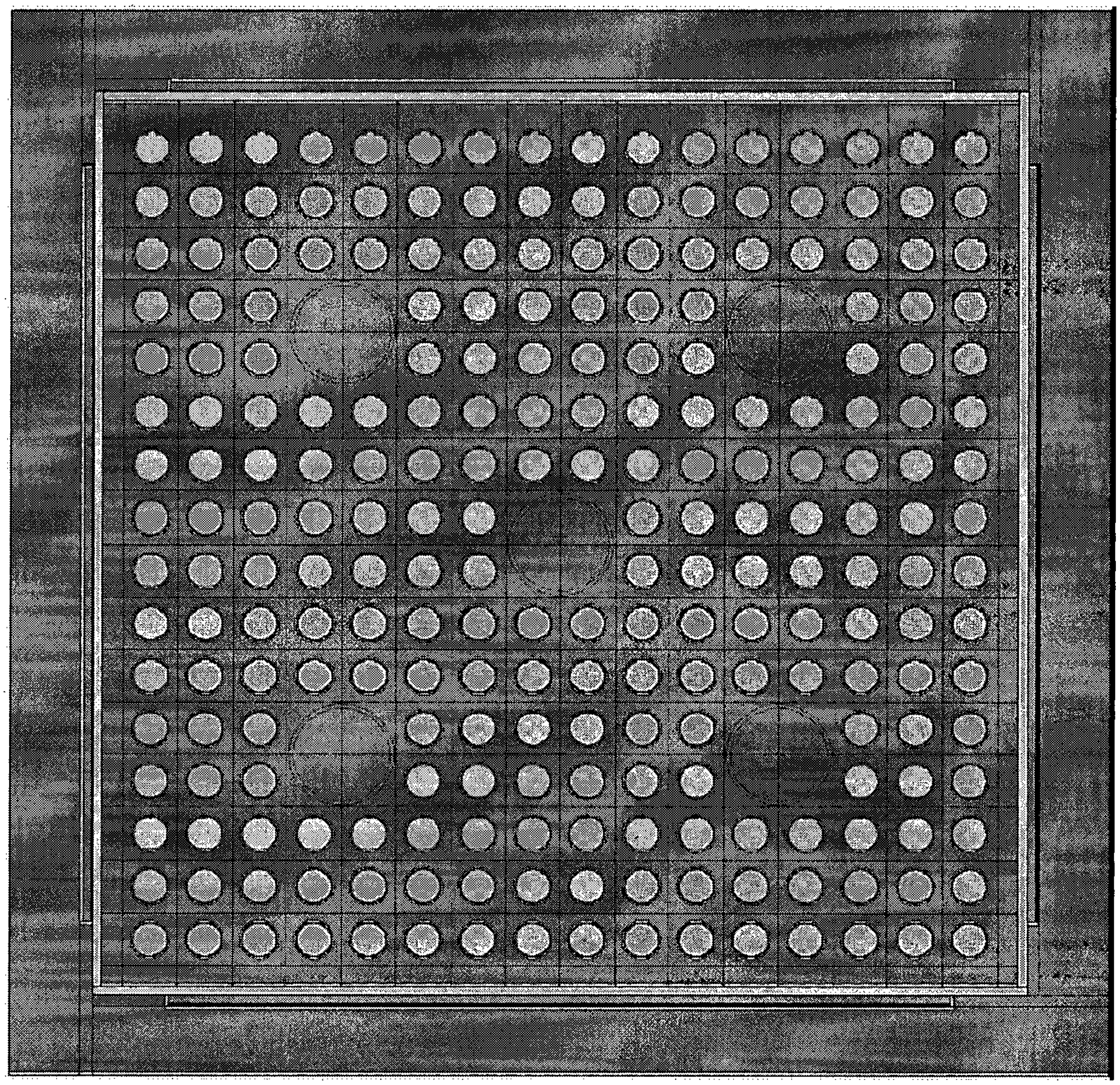


Figure 5.1 Region 1 Model

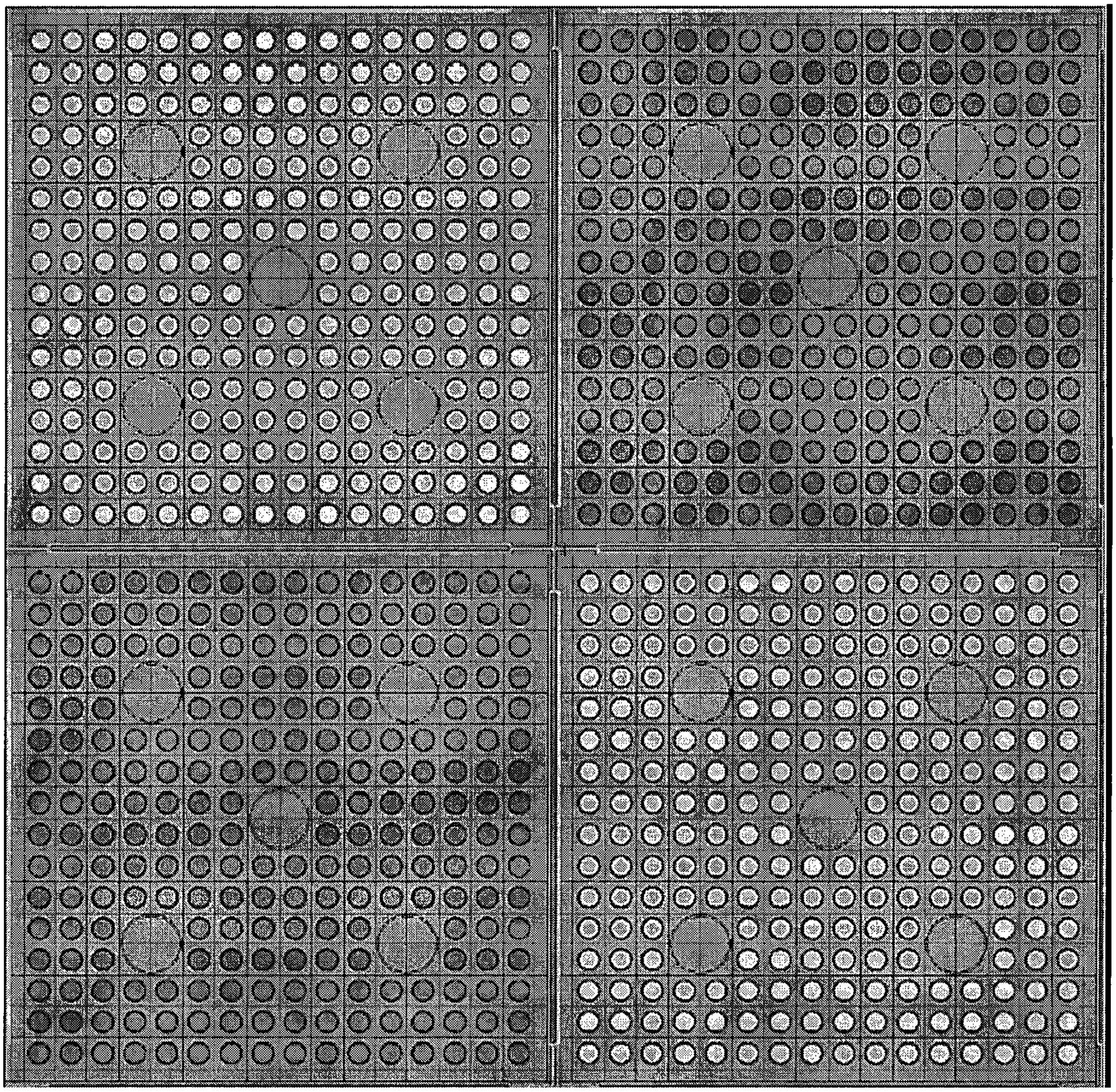


Figure 5.2 Region 2 Model

Figure 7.1
Region 2 Spent Fuel Uniform Loading Burnup versus Enrichment Curve

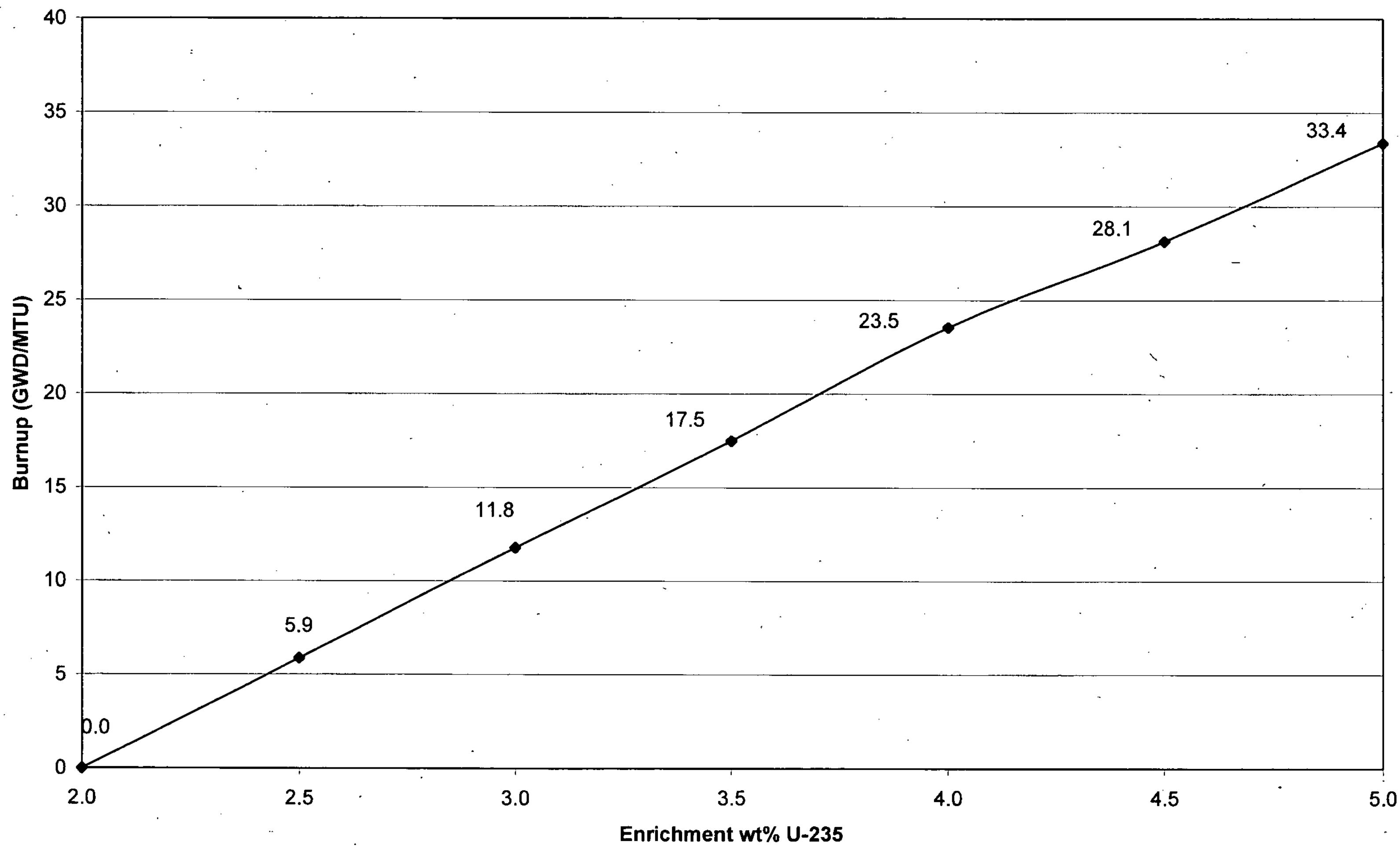


Figure 7.2
Region 2 Spent Fuel Checkerboard Loading Burnup versus Enrichment Curve

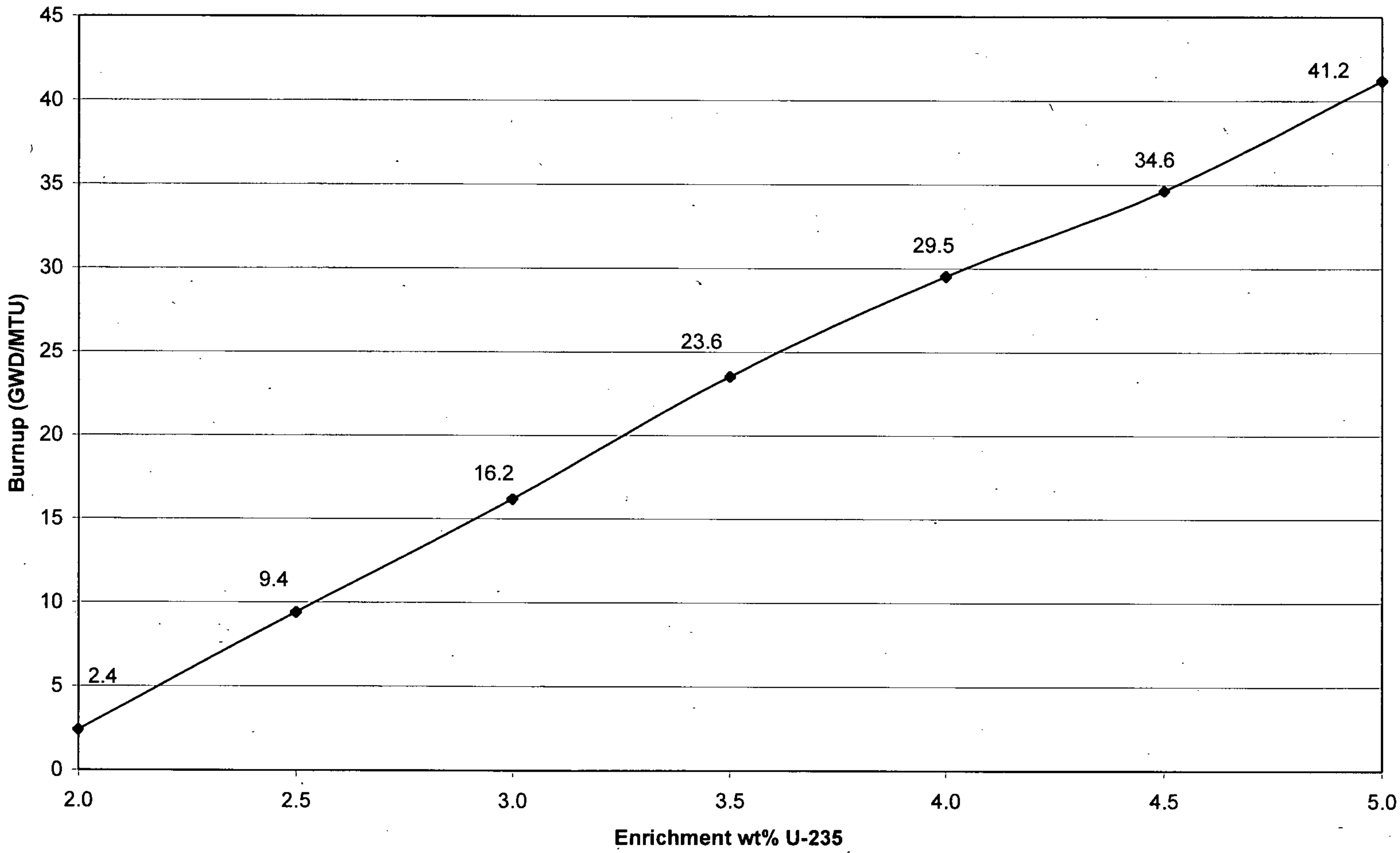
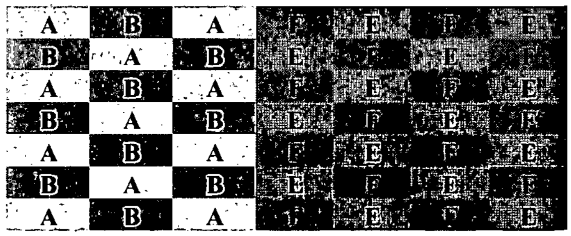
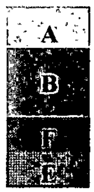


Figure 7.3
Region 2 intra-rack interface between half a rack of Fresh fuel checkerboard and half a rack of spent fuel checkerboard





5 wt% ^{235}U , 27 GWD/MTU
Spent Fuel At Spent Fuel Checkerboard Curve
5 wt% ^{235}U Fresh Fuel
Empty Cell

Figure 7.4
Region 2 intra-rack interface between a 3x3 set of Fresh fuel checkerboard (fresh in center) surrounded by a rack of spent fuel checkerboard

B	A	B	A	B	A	B
A	B	A	B	A	B	A
B	A	F	E	F	A	B
A	B	E	F	E	B	A
B	A	F	E	F	A	B
A	B	A	B	A	B	A
B	A	B	A	B	A	B

A	5 wt% ^{235}U, 27 GWD/MTU
B	Spent Fuel At Checkerboard Curve
F	5 wt% ^{235}U Fresh Fuel
E	Empty Cell

Figure 7.5
Two-Dimensional Representation of the Actual Calculations Model used for the New Fuel Vault as seen from above.

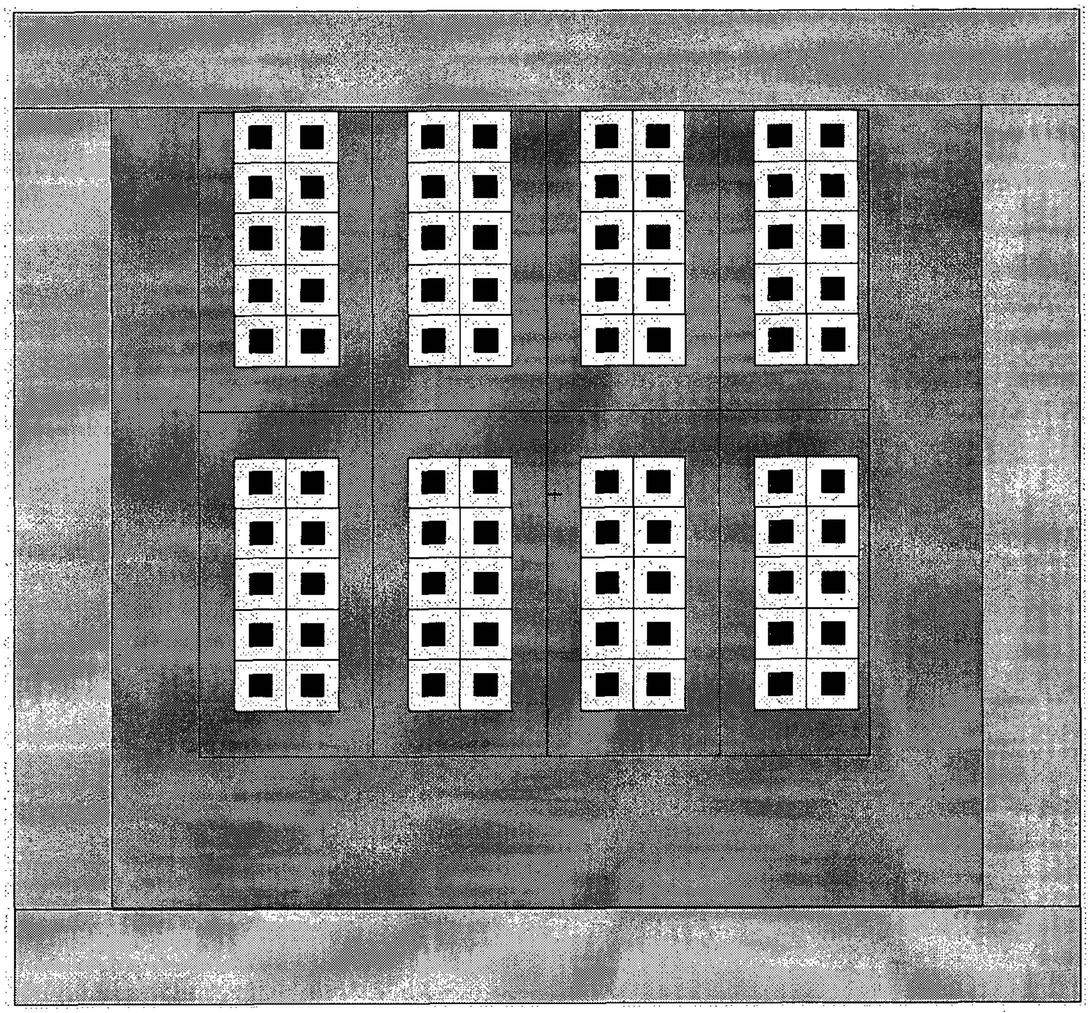


Figure 7.6
Two-Dimensional Representation of the Actual Calculations Model used for the New Fuel Vault as seen from the side.

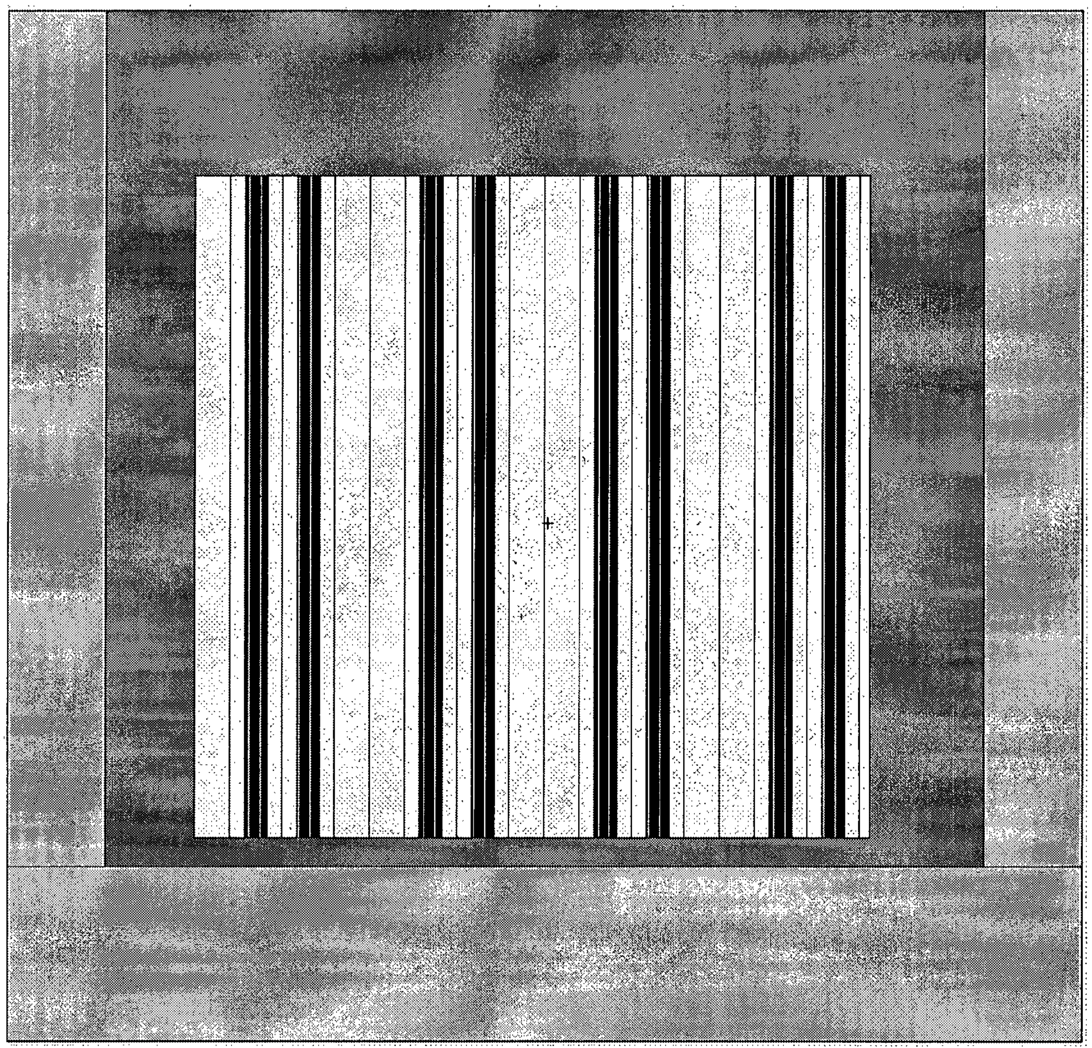
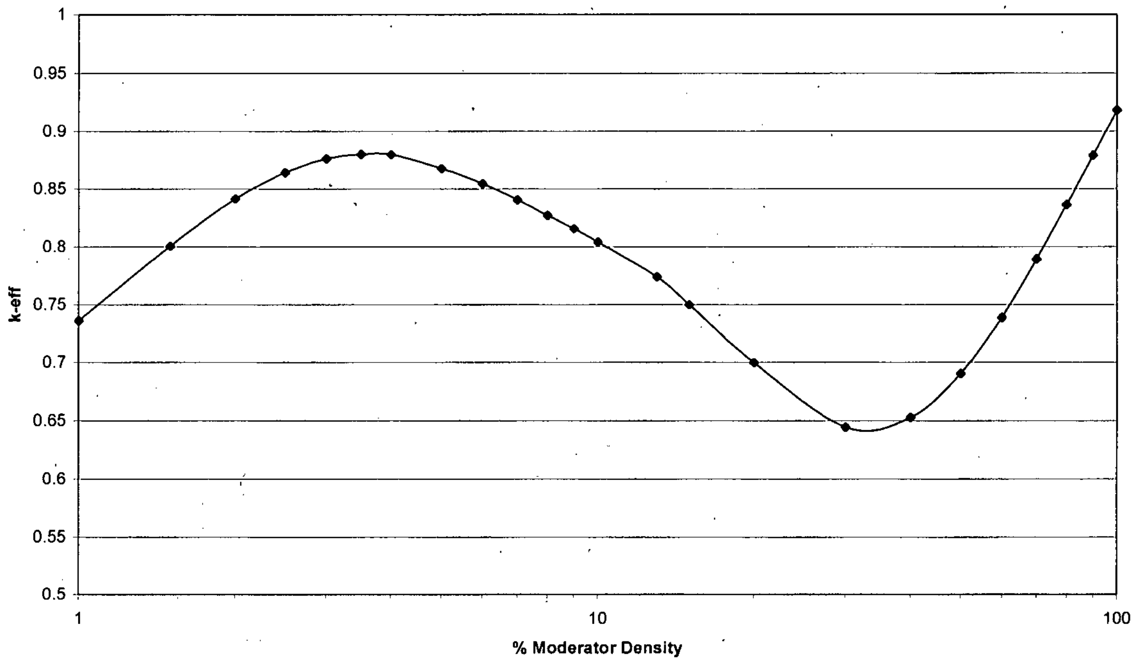


Figure 7.7
Results of the Waterford Unit 3 New Fuel Vault Criticality Analysis As a Function of Water Density



Appendix A
Benchmark Calculations

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HOLTEC PROPRIETARY APPENDIX HAS BEEN REMOVED IN IT'S ENTIRETY