Entergy

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Fred Dacimo
Site Vice President
Administration

July 10, 2006

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-06-020

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station O-P1-17
Washington, DC 20555-0001

Subject: Application for a Technical Specification Change to Add Spent Fuel Cask Loading Requirements

Dear Sir:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) hereby requests an amendment to the Technical Specifications (TS) for Indian Point Unit 2. Specifically, this request proposes to revise Plant Systems Section 3.7 and Design Features Section 4.0 to add technical specification requirements to the 10 CFR 50 license that establish cask storage area boron concentration limits and restrict the minimum burnup of spent fuel assemblies associated with spent fuel cask loading operations. Approval of these changes is needed to support a dry cask storage loading campaign, which Entergy plans to conduct at Indian Point Unit 2 in accordance with the general license provisions of 10 CFR 72, Subpart K, beginning in 2007.

The Nuclear Regulatory Commission (NRC) issued Regulatory Issue Summary (RIS) 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," on March 23, 2005. RIS 2005-05 highlighted differences in the NRC Part 50 criticality requirements for the spent fuel pool and Part 72 requirements for spent fuel storage casks and emphasized that licensees are expected to comply with both Part 50 and Part 72 during cask loading operations. This request is consistent with the regulatory guidance provided in RIS 2005-05.

A new criticality analysis has been performed for Indian Point Unit 2 using a methodology approved previously by the NRC for soluble boron and burnup credits in the IP2 spent fuel pit. The new criticality analysis demonstrates acceptable subcriticality margins for cask loading operations in the cask storage area in accordance with Part 50. Accordingly, new technical specification requirements have been developed that are consistent with those contained in the Indian Point Unit 2 Technical Specifications for spent fuel in the spent fuel storage racks, and are hereby proposed for NRC approval. The Indian Point Unit 2 Part 50 and Part 72 criticality analyses are independent. Therefore, nothing in this submittal is intended to replace or supersede any Part 72 requirement.

A001

Entergy has evaluated the proposed change in accordance with 10 CFR 50.91(a)(1) using the criteria of 10 CFR 50.92(c) and has determined that this proposed change involves no significant hazards considerations, as described in Attachment 1. The proposed changes to the Technical Specifications and Bases are provided in Attachment 2.

Attachment 3 provides the new criticality analysis for soluble boron and burnup credit of the HI-STORM 100 MPC-32 in the Indian Point 2 spent fuel pit. The Appendix to this analysis documents the results of benchmark calculations used to compute the reactivity state of fuel assemblies in close-packed arrays and is included here as Attachment 4.

Entergy requests approval of the proposed license amendments by March 31, 2007, to support the dry cask storage loading campaign. The proposed changes will be implemented prior to loading spent fuel in a spent fuel storage cask.

In accordance with 10 CFR 50.91, a copy of this application, with attachments is being provided to the designated New York State official.

No new regulatory commitments are being made by Entergy in this correspondence.

If you have any questions or require additional information, please contact Mr. Patric W. Conroy, Licensing Manager at 914-734-6668.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 07/10/2006.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

cc: next page

Attachments:

1. Analysis of Proposed Technical Specification Changes Regarding the Addition of Spent Fuel Cask Loading Requirements
2. Marked-Up Technical Specification and Corresponding Bases Pages Regarding the Addition of Spent Fuel Cask Loading Requirements
3. Criticality Analysis for Soluble Boron and Burnup Credit of the HI-STORM 100 Multi-Purpose Canister (MPC-32) in the Indian Point 2 Spent Fuel Pool
4. Benchmarking Computer Codes for Calculating the Reactivity State of Spent Fuel Storage Racks, Storage Casks and Transportation Casks

cc:

Mr. John P. Boska, Senior Project Manager
Project Directorate I,
Division of Licensing Project Management
U.S. Nuclear Regulatory Commission

Regional Administrator
Region I
U.S. Nuclear Regulatory Commission

Resident Inspector's Office
IP2

Mr. Paul Eddy
NYS Department of Public Service

ATTACHMENT 1 TO NL-06-020

ANALYSIS OF PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING THE ADDITION OF SPENT FUEL CASK LOADING REQUIREMENTS

Entergy Nuclear Operations, Inc.
Indian Point Unit No. 2
Docket No. 50-247

1.0 DESCRIPTION

This letter is a request to amend Operating License No. DPR-26 for Indian Point Unit 2 (IP2).

Entergy is planning to operate an Independent Spent Fuel Storage Installation (ISFSI) facility at the Indian Point Energy Center in accordance with the general license provisions of 10 CFR 72, Subpart K, using the Holtec HI-STORM 100 Cask System Multi-Purpose Canister (MPC-32). To support this activity, this request proposes to revise the Indian Point Unit 2 Technical Specifications, Section 3.7, Plant Systems, including corresponding Bases, and Section 4.0, Design Features, to add technical specification requirements that establish cask storage area boron concentration limits for spent fuel cask operations, and that restrict the minimum burnup of spent fuel assemblies that can be loaded into a spent fuel cask. These proposed spent fuel cask loading operation requirements are consistent with those contained in the Indian Point Unit 2 (IP2) Technical Specifications for spent fuel in the spent fuel storage racks.

2.0 PROPOSED CHANGES

The following changes are proposed to the IP2 Technical Specifications (TS):

1. A new TS Limiting Condition for Operation (LCO) will be added, LCO 3.7.15, to identify that there is a cask storage area boron concentration loading restriction for the MPC-32 storage cask. Associated with this LCO is a new proposed Bases B 3.7.15.

2. A new Surveillance Requirement (SR), SR 3.7.15.1, will be added to verify that the cask storage area boron concentration is within limit at the specified frequency.

3. A new TS Limiting Condition for Operation will be added, LCO 3.7.16, to identify that there are fuel assembly initial enrichment and burnup loading restrictions for the MPC-32 storage cask included in TS Figure 3.7.16-1. Associated with this LCO is a new proposed Bases B 3.7.16.

4. A new Surveillance Requirement, SR 3.7.16.1, will be added to verify that the fuel assemblies are placed in a MPC-32 storage cask within the limits of Figure 3.7.16-1.

5. A new TS Figure, Figure 3.7.16-1 will be added that specifies acceptable combinations of fuel assembly initial enrichment, burnup and cooling times for MPC-32 cask loading operations.

6. A new Design Feature will be added, 4.3.1.3, to identify that the MPC-32 storage casks are designed to and shall be maintained within certain limits.

In summary, the IP2 TSs will be modified to govern the fuel loading restrictions related to the Holtec HI-STORM 100 MPC-32 storage cask. Following approval of the proposed change, Entergy will make changes to the IP2 UFSAR and TS Bases to appropriately reflect the criticality analysis that was performed for the MPC-32 storage cask.

3.0 BACKGROUND

The spent fuel storage pit is described in Section 9.5.2.1.4 of the UFSAR. It is designed for the underwater storage of spent fuel assemblies, failed fuel cans if required, and control rods/inserts after their removal from the reactor. Storage racks are provided to hold spent fuel

assemblies and are erected on the pit floor. Fuel assemblies are held in a square array, and placed in vertical cells. Fuel inserts are stored in place inside the spent fuel assemblies. An area of the pit is set aside for the placement of a spent fuel cask. The same equipment and procedural controls for controlling fuel within the Spent Fuel Pit (SFP) are utilized when loading/unloading a storage cask.

The installation of a new 110-ton Fuel Handling Building gantry crane will facilitate handling of the transfer cask for the Holtec HI-STORM cask system. The crane main hoist has a 100-ton capacity, while an auxiliary hoist has a 45-ton capacity. Each hoist meets the single failure proof requirements of NUREG-0554 "Single-Failure-Proof Cranes for Nuclear Power Plants". The crane will be used to move dry cask storage equipment into and out of the spent fuel pit. The license amendment request regarding the new gantry crane was submitted for NRC approval in Reference 1. NRC approval of the request is documented in Reference 2.

The NRC criteria for criticality control during spent fuel cask loading operations have been historically governed by the requirements of 10 CFR 72. Similarly, the criteria for criticality control of spent fuel stored in the spent fuel pit storage racks are governed by the requirements of 10 CFR 50. Part 50 and Part 72 have different acceptance criteria that provide adequate assurance that the spent fuel will remain subcritical. The Indian Point Unit 2 Part 50 and Part 72 criticality analyses are independent. Therefore, nothing in this submittal is intended to replace or supersede any Part 72 requirement.

Part 50 requires that spent fuel in the spent fuel storage racks remain subcritical (i.e., $k_{eff} < 1.0$) when fully flooded with unborated water (i.e., boron dilution event). In order to maintain $k_{eff} < 1.0$ when flooded with unborated water, the NRC allows licensees to credit the reduced reactivity of the spent fuel associated with burnup during operation. However, Part 72 requires that all fuel in the cask be considered to be fresh fuel at the maximum enrichment allowed by the Certificate of Compliance (CoC) for the spent fuel cask system. As a result, Part 72 requires soluble boron credit to maintain spent fuel in the cask subcritical during cask loading operations. These differences, and the need to comply with both Part 50 and Part 72 during cask loading operations, are described in RIS 2005-05 (Ref. 3). In addition, the minimum soluble boron concentration required by Part 50 and Part 72 is also impacted by differences in the geometry and credit allowed for the performance of the fixed neutron absorber in the spent fuel storage racks and spent fuel cask as specified in the governing Technical Specifications.

Consequently, Entergy determined that a new Part 50 criticality analysis was needed for IP2 to demonstrate acceptable subcriticality margins for cask loading operations in the cask storage area given a boron dilution event, consistent with RIS 2005-05. Therefore, a new analysis was performed, as provided in Attachment 3, using the same methodology as previously approved by the NRC for IP2 for spent fuel rack storage (Refs. 4, 5, and 6). The analysis provided in Attachment 3 demonstrates acceptable subcriticality margins for IP2 cask loading operations and postulated cask loading events.

4.0 TECHNICAL ANALYSIS

The original spent fuel racks (SFRs) were replaced with new SFRs in 1990 to increase the on-site storage capacity for spent fuel, as discussed in Reference 4. The capacity of the pit was increased by decreasing the spacing between adjacent fuel assemblies. This decreased spacing was compensated for by using Boraflex neutron absorbers between rack cells in order to maintain a sufficiently subcritical configuration. Since Boraflex is susceptible to in-service

degradation, a RACKLIFE computer model of the IP2 SFP was developed. The RACKLIFE model provides a means for predicting the rate at which each panel of Boraflex accumulates gamma exposure, and therefore provides a means for evaluating and implementing rack management strategies to mitigate the effects of Boraflex degradation. The RACKLIFE analysis indicated that areas of moderate dissolution of the Boraflex panels had likely occurred. BADGER (Boron-10 Areal Density Gage for Evaluating Racks) tests performed initially in 2000 and then again in 2003 confirmed the predictions of the RACKLIFE model. In order to offset the reactivity effects of the degraded Boraflex, criticality analyses, as described in Reference 4, were performed in 2001 to take credit for soluble boron and burnup. The analysis showed that sufficient subcritical margin could be maintained through 2006. In order to confirm the continuing applicability of the criticality analyses beyond 2006 the RACKLIFE model will be updated to include those fuel moves made during the Spring 2006 refueling outage. BADGER tests are currently scheduled for Summer 2006 to confirm the RACKLIFE predictions. A more detailed discussion of RACKLIFE and BADGER testing is provided in Reference 4.

In support of this amendment request, a new criticality analysis was performed using the same methodology previously approved by the NRC for soluble boron and burnup credits in the IP2 spent fuel pit (Refs. 4, 5, and 6). The analysis documented in Attachment 3 was performed for a fully loaded, thirty-two assembly, multi-purpose canister (MPC-32) and transfer cask positioned in the cask pit in the southwest corner of the IP2 spent fuel pit. It was determined that under conditions of maximum reactivity, when loaded with fuel of the maximum allowable enrichment (1.8 w/o U-235 at zero burnup or up to 5.0 w/o U-235 with credit for burnup and Integral Fuel Burnable Absorbers (IFBAs)), that k_{eff} is less than 1.0 without credit for soluble boron during cask loading operations. In addition, it was determined that under the same assumptions, a soluble boron concentration of 250 ppm will maintain $k_{eff} \leq 0.95$ during the loading of fuel assemblies into the MPC-32, while in the spent fuel pit. Three off-normal accident conditions (i.e. misloading a fresh fuel assembly into the MPC, placement of a fully loaded MPC-32 adjacent to the most reactive spent fuel assemblies or the accidental dropping of the maximum reactivity fuel assembly onto the fully loaded MPC-32) were evaluated. For the bounding case (misloading a fresh fuel assembly into the MPC), an additional 121 ppm of soluble boron (a total soluble boron concentration of 371 ppm) is required to maintain $k_{eff} \leq 0.95$.

Boron dilution analyses previously performed, as documented in Reference 4, determined the dilution volumes required to dilute the spent fuel pit from 2000 ppm to 786 ppm soluble boron, the lower value being the concentration required to maintain $k_{eff} \leq 0.95$ for spent fuel stored in the spent fuel storage racks. The concentration of 786 ppm includes soluble boron credit for uncertainties related to burnup, and is more than the amount required (250 ppm) to reduce k_{eff} by 0.05 for spent fuel in the MPC-32 storage cask for a dilution event.

Administrative procedures currently in place will prevent a dilution from occurring, which could potentially reduce the spent fuel pit boron concentration to a value that would result in k_{eff} being greater than 0.95. For the present analysis, the soluble boron concentration was assumed to be only that required to reduce k_{eff} by 0.05, therefore the boron dilution analysis remains valid and it is conservative with respect to dilution volumes and times.

The criticality analysis provided in Attachment 3 provides the basis for the proposed Technical Specification changes necessary for cask loading operations to be consistent with the Part 50 licensing bases for spent fuel in the spent fuel storage racks. The results demonstrate that the

spent fuel pit boron concentration limits associated with the existing Part 50 boron dilution analysis for the spent fuel storage racks remain bounding for cask loading operations. For the purposes of addressing spent fuel pit boron dilution events, the cask storage area is considered part of the spent fuel pit volume. Accordingly, k_{eff} will be ≤ 0.95 in the spent fuel cask in the event the spent fuel pit is flooded with borated water to 786 ppm. Additionally, k_{eff} will be < 1.0 in the spent fuel cask in the event the spent fuel pit is flooded with unborated water.

Criticality considerations associated with postulated fuel mishandling events during cask loading operations were included in the scope of the new criticality analysis provided in Attachment 3. The required boron concentration necessary to mitigate the most severe event was determined to be 371 ppm, as indicated above. This is well below the existing LCO 3.7.12 limit of 2000 ppm for the spent fuel pit and the LCO limit of 2000 ppm in proposed LCO 3.7.15 for cask loading operations provided in Attachment 2.

Derived from the new criticality analysis are new technical specification requirements for cask loading operations in the cask storage area, as provided in Attachment 2. Proposed TS 3.7.15 is a new Plant Systems specification that establishes a boron concentration limit of ≥ 2000 ppm for the cask storage area. Additionally, proposed TS 3.7.16 is a new Plant Systems specification that incorporates a family of burnup versus enrichment curves, for various cooling times, that establishes minimum fuel burnup limits for spent fuel that can be stored in a cask. Consistent with these changes, proposed TS 4.3.1.3 has also been added to the Design Features specifications that establish design requirements to support cask loading operations.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

In 10 CFR 50.92(c), the NRC provides the following standards to be used in determining the existence of a Significant Hazards Consideration:

".. a proposed amendment to an operating license for a facility licensed under 10 CFR 50.21(b) or 10 CFR 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety."

Entergy has reviewed the proposed license amendment request and has determined that its adoption does not involve a Significant Hazards Consideration as discussed below:

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

   Response: No.

The proposed change will revise the Indian Point Unit 2 Technical Specifications associated with the SFP to assure that the regulatory requirements related to criticality in the SFP and applied to the Holtec HI-STORM 100 Multi-Purpose Canister MPC-32 when in the SFP are reflected in the IP2 TS. The proposed change does not require any physical changes to Part 50 structures, systems, or components, nor will their performance requirements be altered.

Therefore, the response of the plant to previously analyzed accidents and related radiological releases will not be adversely impacted, and will bound those postulated during cask loading activities in the cask storage area. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

Existing fuel handling procedures and associated administrative controls remain applicable for cask loading operations within the SFP. Additionally, the soluble boron concentration required to maintain $k_{eff} \leq 0.95$ for postulated criticality accidents associated with cask loading operations was also evaluated. The results of the analyses, using a methodology previously approved by the NRC, demonstrate that the amount of soluble boron required to compensate for the positive reactivity associated with these postulated accidents (371 ppm) remains well below the existing spent fuel pit minimum boron concentration limit of 2000 ppm. Accordingly, the same limit has been proposed for cask loading operations in the cask storage area. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

Response: No.

An NRC approved methodology was used to perform the criticality analysis which provides the basis to incorporate a new family of burnup versus enrichment curves, for various cooling times, into the plant Technical Specifications to ensure criticality requirements are met during spent fuel cask loading. Accordingly, the existing minimum boron concentration limit for the spent fuel pit of 2000 ppm will continue to remain bounding during cask loading operations. This determination accounts for uncertainties at a 95 percent probability, 95 percent confidence level. Should it be postulated that a boron dilution event does occur during this time period, k_{eff} will remain less than 1.0 should the cask storage area become fully flooded with unborated water. Therefore, there will not be a significant reduction in a margin of safety.

Based upon the preceding information, Entergy has concluded that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

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

Because no design changes are associated with this amendment request, the following 10 CFR 50, Appendix A, General Design Criteria (GDC) remain satisfied as described in the IP2 Updated Final Safety Analysis Report:

- GDC 66 – Prevention of Fuel Storage Criticality

- GDC 67 – Fuel and Waste Storage Decay Heat
- GDC 68 – Fuel and Waste Storage Radiation Shielding
- GDC 69 – Protection Against Radioactivity Release from Spent Fuel and Waste Storage

The existing Indian Point Unit 2 Technical Specifications govern the spent fuel pit boron concentration, the maximum U-235 fuel enrichment that can be stored in the SFP, and the loading restrictions based on cooling time, initial fuel enrichment, IFBA loading, and fuel burnup. In addition, the existing IP2 TSs govern the criticality requirements, which include maintaining the effective multiplication factor (k_{eff}) less than or equal to 0.95 associated with fuel stored in the SFP. Spent fuel pit loading is also governed by 10 CFR 50.68, *Criticality Accident Requirements*. Criticality evaluations are performed for spent fuel that will be stored in the SFP based on the requirements set forth in 10 CFR 50.68.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, and do not affect conformance with any GDC differently than described in the UFSAR. Entergy has determined that a TS change is appropriate to support loading/unloading an MPC-32.

5.3 Environmental Considerations

Entergy has evaluated the proposed changes and determined the changes do not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or (3) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criteria 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 previously approved similar applications regarding cask loading operations in the cask storage area analyzed in accordance with 10 CFR 50 requirements. An example of such an approval is that granted to Farley Units 1 and 2 dated June 28, 2005.

7.0 REFERENCES

1. Letter NL-04-126, Entergy to USNRC, "License Amendment Request (LAR) – Fuel Storage Building Single-Failure-Proof Gantry Crane," dated November 1, 2004.

2. NRC Safety Evaluation Report, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment Re: Approving the Use of a New Gantry Crane in the Fuel Storage Building," dated November 21, 2005.

3. Regulatory Issue Summary (RIS) 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installation," dated March 23, 2005.

4. Letter NL-01-110, Entergy to USNRC, "License Amendment Request (LAR 01-010) for Spent Fuel Storage Pit Rack Criticality Analysis with Soluble Boron Credit," dated September 20, 2001.

5. Letter NL-02-013, Entergy to USNRC, "Response to Request for Additional Information Regarding Spent Fuel Storage Pit Analysis with Soluble Boron Credit, Indian Point Nuclear Generating Unit No. 2," dated January 25, 2002.

6. NRC Safety Evaluation Report, "Indian Point Nuclear Generating Unit No. 2 – Amendment Re: Credit for Soluble Boron and Burnup in Spent Fuel Pit," dated May 29, 2002.

ATTACHMENT 2 TO NL-06-020

MARKED-UP TECHNICAL SPECIFICATION AND CORRESPONDING BASES PAGES REGARDING THE ADDITION OF SPENT FUEL CASK LOADING REQUIREMENTS

Technical Specification pages:

Page iii
Page 3.7.15-1
Page 3.7.15-2
Page 3.7.16-1
Page 3.7.16-2
Page 4.0-2

Technical Specification Bases pages (for information only):

Page ii
Pages B 3.7.15-1,-2,-3,-4, and -5
Pages B 3.7.16-1,-2,-3 and -4

Entergy Nuclear Operations, Inc.
Indian Point Unit No. 2
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PROPOSED NEW SPECIFICATION

3.7 PLANT SYSTEMS

3.7.15 Cask Storage Area Boron Concentration --- Cask Loading Operations

LCO 3.7.15 The cask storage area boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the cask storage area.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Cask storage area boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	A.1 Suspend movement of fuel assemblies in the cask storage area.	Immediately
	AND	
	A.2 Initiate action to restore cask storage area boron concentration to within limit.	Immediately

PROPOSED NEW SPECIFICATION

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the cask storage area boron concentration is within limit.	Once within 4 hours prior to entering the Applicability of this LCO. AND Every 48 hours thereafter.

PROPOSED NEW SPECIFICATION

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Assembly Storage --- Cask Loading Operations

LCO 3.7.16 The combination of initial enrichment and burnup of each spent fuel assembly stored in the cask storage area shall be within the acceptable burnup domain of Figure 3.7.16-1.

APPLICABILITY: Whenever any fuel assembly is stored in the cask storage area.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1.	Prior to placing fuel assemblies in the spent fuel cask.

PROPOSED NEW SPECIFICATION

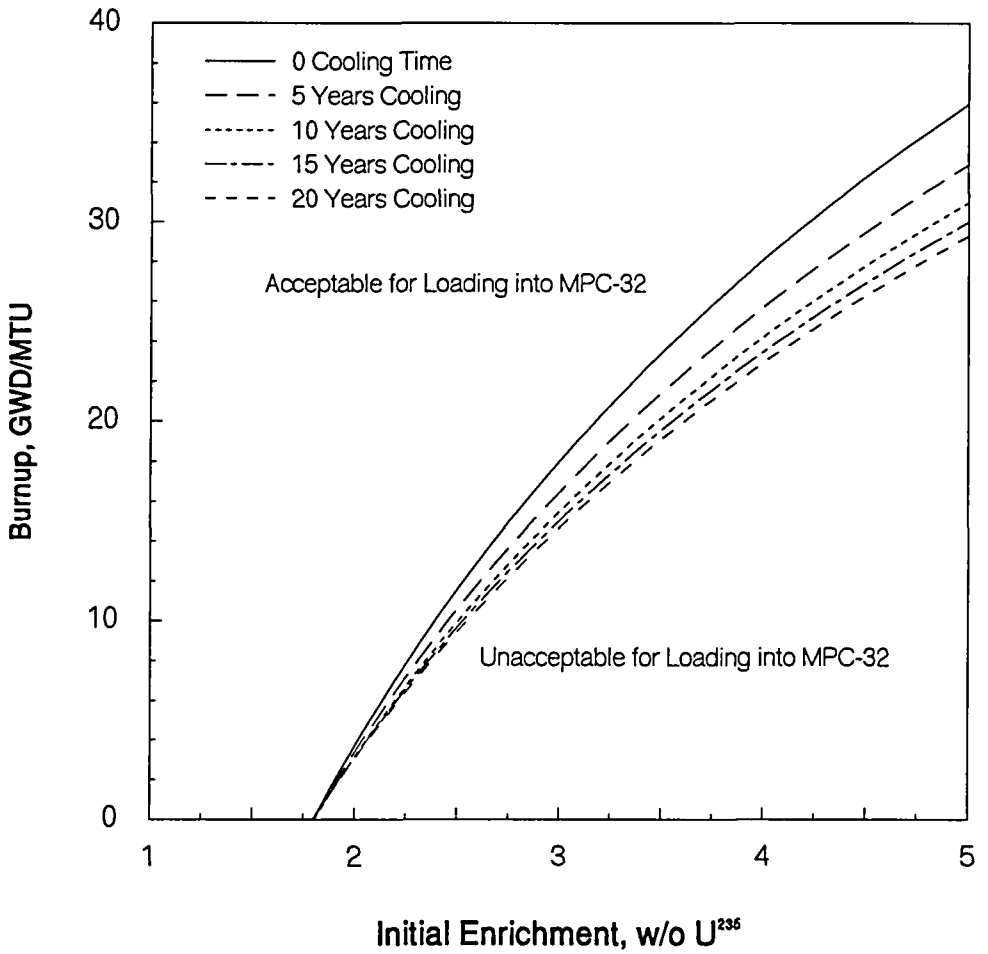

Figure 3.7.16-1
Fuel Assembly Burnup Limit Requirements for Cask Storage

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. $k_{eff} < 1.0$ if fully flooded with unborated water, and

c. Each fuel assembly classified based on initial enrichment, burnup, cooling time and number of Integral Fuel Burnable Absorbers (IFBA) rods with individual fuel assembly storage location within the spent fuel storage rack restricted as required by Technical Specification 3.7.13.

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent, and poisons, if necessary, to meet the limit for k_{eff},

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, and

c. A 20.5 inch center to center distance between fuel assemblies placed in the storage racks to meet the limit for k_{eff}.

Insert 4.3.1.3

4.3.2 Drainage

The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pit below a nominal elevation of 88 feet, 6 inches.

4.3.3 Capacity

The spent fuel pit is designed and shall be maintained with a storage capacity limited to no more than 269 fuel assemblies in Region I and 1105 fuel assemblies in Region II.

Insert 4.3.1.3

4.3.1.3 The spent fuel casks are designed and shall be maintained with:

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

b. $k_{eff} < 1.0$ if fully flooded with unborated water;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 250 ppm;

d. A nominal 9.218 inch center to center distance between fuel assemblies placed in the spent fuel cask; and

e. Spent fuel assemblies with a combination of discharge burnup and initial enrichment in the acceptable burnup domain of Figure 3.7.16-1.

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PROPOSED NEW SPECIFICATION BASES

B 3.7 PLANT SYSTEMS

B 3.7.15 Cask Storage Area Boron Concentration --- Cask Loading Operations

BASES

BACKGROUND

The cask storage area is located in the southwest corner of the spent fuel pit and is used to facilitate cask loading operations. The spent fuel cask contains storage locations for 32 fuel assemblies. Westinghouse 15 x 15 fuel assemblies with initial enrichments less than or equal to 5.0 weight percent U-235 can be stored in the spent fuel cask provided the fuel burnup-enrichment combinations are within the limits specified in Figure 3.7.16-1 of the Technical Specifications. Northeast Technology Corp. Report NET-239-02, "Criticality Analysis for Soluble Boron and Burnup Credit of the HI-STORM 100 Multi-Purpose Canister (MPC-32) in the Indian Point Unit 2 Spent Fuel Pool" (Ref. 4) provides the basis for acceptability to conduct cask loading operations in the cask storage area.

The above methodology ensures that the spent fuel cask multiplication factor, k_{eff}, is less than or equal to 0.95, as recommended by ANSI 57.2-1983 (Ref. 3) and NRC Guidance (Refs. 1, 2 and 6). A storage configuration is defined using k_{eff} calculations to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95. The treatment of reactivity uncertainties, as well as the calculation of postulated accidents crediting soluble boron is described in Reference 4.

The above methodology was used to evaluate cask loading of Westinghouse 15 x 15 fuel assemblies with initial enrichments less than or equal to 5.0 weight percent U-235 in the spent fuel cask during loading operations in the cask storage area. The resulting enrichment and burnup limits are shown in Figure 3.7.16-1.

A cask storage area boron concentration of 2000 ppm ensures that no credible boron dilution event will result in a k_{eff} greater than 0.95.

PROPOSED NEW SPECIFICATION BASES

BASES

APPLICABLE SAFETY ANALYSES

The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under accident conditions was determined by evaluating all credible events which increase the k_{eff} value of the spent fuel cask (Ref. 4). The accident event which produces the largest increase in the spent fuel cask k_{eff} value is employed to determine the required soluble boron concentration necessary to mitigate this and all less severe accident events. The list of accident cases considered includes:

- Dropped fresh fuel assembly on top of the spent fuel cask,
- Dropped fresh fuel assembly outside of the spent fuel cask,
- Spent fuel cask assembly-to-assembly pitch reduction due to seismic event,
- Spent fuel cask water temperature greater than 39°F but less than or equal to 212°F,
- Misloaded fresh fuel assembly into a spent fuel cask location.

It is possible to drop a fuel assembly on top, or immediately outside, of the spent fuel cask. In this case, the physical separation (approximately 20 inches) between the fuel assemblies loaded inside the spent fuel cask and the assembly lying on top or outside is sufficient to neutronically decouple the accident. A bundle dropped in the northeast corner of the cask storage area adjacent to the spent fuel storage racks will produce a very small positive reactivity increase. This small increase will not be as limiting as the reactivity increase associated with a fuel misloading event inside the spent fuel cask.

For the accident due to a seismic event, the assembly-to-assembly pitch is reduced relative to the reference case with all bundles centered in the storage basket locations. A slight decrease of k_{eff} (but within the statistical uncertainty) is determined for this case, and this is significantly less than the reactivity increase due to a fuel misloading event inside the spent fuel cask.

The nominal water temperature range addressed for the spent fuel cask in this analysis is greater than 39°F but less than or equal to 212°F. The reference case (68°F) assumes maximum water density (1.0 gram/cc). An increase in moderator temperature results in a decrease in reactivity. Therefore, at higher temperatures, the fuel misloading event remains limiting.

PROPOSED NEW SPECIFICATION BASES

BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel assembly misloading accident represents the most severe postulated event for reactivity insertion and involves the placement of a fresh fuel assembly into a spent fuel cask location. For the limiting case, a fresh fuel assembly misloaded into a central storage cell, an additional 121 ppm of soluble boron is required to maintain $k_{eff} \leq 0.95$. The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under normal operating conditions is 250 ppm, therefore, a total soluble boron concentration of 371 ppm is required to accommodate the limiting accident. This is well below the LCO limit of 2000 ppm.

As described in the Bases for LCO 3.7.13, a spent fuel pit boron dilution evaluation (Ref. 5) determined that the volume of water necessary to dilute the spent fuel pit from the LCO limit of 2000 ppm to 786 ppm (the boron concentration required to maintain k_{eff} less than or equal to 0.95) is approximately 230,551 gallons (Ref. 7). A spent fuel pit dilution of this volume is not a credible event, since it would require this large volume of water to be transferred from a source to the spent fuel pit, ultimately overflowing the pit. This event would be detected and terminated by plant personnel prior to exceeding a k_{eff} of 0.95.

The concentration of dissolved boron in the cask storage area satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The cask storage area boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pit preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in References 4 and 5. The specified boron concentration of 2000 ppm ensures that the spent fuel cask k_{eff} will remain less than or equal to 0.95 due to a postulated fuel assembly misloading accident (371 ppm) or boron dilution event (250 ppm) for the MPC-32.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the cask storage area of the spent fuel pit.

PROPOSED NEW SPECIFICATION BASES

BASES

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

If the LCO is not met in MODE 5 or 6, LCO 3.0.3 would not be applicable. Moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, or restoring cask storage area boron concentration is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies or restore boron concentration, is not sufficient reason to require a reactor shutdown.

When the concentration of boron in the cask storage area of the fuel storage pit is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. Action is also initiated to restore the concentration of boron simultaneously with suspending movement of fuel assemblies.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

The boron concentration in the spent fuel cask area water must be verified to be within limit within four hours prior to entering the Applicability of the LCO. For loading operations, this means within four hours of loading the first fuel assembly into the cask.

For unloading operations, this means verifying the concentration of the borated water source to be used to re-flood the spent fuel cask within four hours of commencing re-flooding operations. This ensures that when the LCO is applicable (upon introducing water into the spent fuel cask), the LCO will be met.

The frequency of every 48 hours thereafter applies if cask loading operations continue for 48 hours or more and continue until the spent fuel cask is removed from the cask storage area.

PROPOSED NEW SPECIFICATION BASES

BASES

REFERENCES

1. USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, June, 1987.

2. USNRC Spent Fuel Storage Facility Design Bases (for Comment) Proposed Revision 2, 1981.

3. ANS, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," ANSI/ANS-57.2-1983.

4. NET-239-02, "Criticality Analysis for Soluble Boron and Burnup Credit of the HI-STORM 100 Multi-Purpose Canister (MPC-32) in the Indian Point Unit 2 Spent Fuel Pool", Revision 2, Northeast Technology Corp.; Kingston, NY, 20 April 2006.

5. Indian Point 2 UFSAR, Section 14.2.1.

6. NRC, Letter to all Power Reactor Licensees from B.K. Grimes "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

7. NET-173-02, "Indian Point Unit 2 Spent Fuel Pool (SFP) Boron Dilution Analysis", Revision 1, Northeast Technology Corp.; Kingston, NY, 12 September 2001.

PROPOSED NEW SPECIFICATION BASES

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Assembly Storage --- Cask Loading Operations

BASES

BACKGROUND

The cask storage area is located in the southwest corner of the spent fuel pit and is used to facilitate cask loading operations. The spent fuel cask contains storage locations for 32 fuel assemblies. Westinghouse 15 x 15 fuel assemblies with initial enrichments less than or equal to 5.0 weight percent U-235 can be stored in the spent fuel cask provided the fuel burnup-enrichment combinations are within the limits specified in Figure 3.7.16-1 of the Technical Specifications. Northeast Technology Corp. Report NET-239-02, "Criticality Analysis for Soluble Boron and Burnup Credit of the HI-STORM 100 Multi-Purpose Canister (MPC-32) in the Indian Point Unit 2 Spent Fuel Pool" (Ref. 4) provides the basis for acceptability to conduct cask loading operations in the cask storage area.

The above methodology ensures that the spent fuel cask multiplication factor, k_{eff}, is less than or equal to 0.95, as recommended by ANSI 57.2-1983 (Ref. 3) and NRC Guidance (Refs. 1, 2 and 5). A storage configuration is defined using k_{eff} calculations to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95. The treatment of reactivity uncertainties, as well as the calculation of postulated accidents crediting soluble boron is described in Reference 4.

The above methodology was used to evaluate cask loading of Westinghouse 15 x 15 fuel assemblies with initial enrichments less than or equal to 5.0 weight percent U-235 in the spent fuel cask during loading operations in the cask storage area. The resulting enrichment and burnup limits are shown in Figure 3.7.16-1.

Westinghouse 15 x 15 fuel assemblies with initial enrichments less than or equal to 5.0 weight percent U-235 can be stored in the spent fuel cask. The fuel assemblies must satisfy the minimum burnup requirement as shown in Figure 3.7.16-1.

PROPOSED NEW SPECIFICATION BASES

BASES

APPLICABLE SAFETY ANALYSES

The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under accident conditions was determined by evaluating all credible events which increase the k_{eff} value of the spent fuel cask (Ref. 4). The accident event which produces the largest increase in the spent fuel cask k_{eff} value is employed to determine the required soluble boron concentration necessary to mitigate this and all less severe accident events. The list of accident cases considered includes:

- Dropped fresh fuel assembly on top of the spent fuel cask,
- Dropped fresh fuel assembly outside of the spent fuel cask,
- Spent fuel cask assembly-to-assembly pitch reduction due to seismic event,
- Spent fuel cask water temperature greater than 39°F but less than or equal to 212°F,
- Misloaded fresh fuel assembly into a spent fuel cask location.

It is possible to drop a fuel assembly on top, or immediately outside of the spent fuel cask. In this case, the physical separation (approximately 20 inches) between the fuel assemblies loaded inside the spent fuel cask and the assembly lying on top or outside is sufficient to neutronically decouple the accident. A bundle dropped in the northeast corner of the cask storage area adjacent to the spent fuel storage racks will produce a very small positive reactivity increase. This small increase will not be as limiting as the reactivity increase associated with a fuel misloading event inside the spent fuel cask.

For the accident due to a seismic event, the assembly-to-assembly pitch is reduced relative to the reference case with all bundles centered in the storage basket locations. A slight decrease of k_{eff} (but within the statistical uncertainty) is determined for this case, and this is significantly less than the reactivity increase due to a fuel misloading event inside the spent fuel cask.

The nominal water temperature range addressed for the spent fuel cask in this analysis is greater than 39°F but less than or equal to 212°F. The reference case (68°F) assumes maximum water density (1.0 gram/cc). An increase in moderator temperature results in a decrease in reactivity. Therefore, at higher temperatures, the fuel misloading event remains limiting.

PROPOSED NEW SPECIFICATION BASES

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The fuel assembly misloading accident represents the most severe postulated event for reactivity insertion and involves the placement of a fresh fuel assembly into a spent fuel cask location. For the limiting case, a fresh fuel assembly misloaded into a central storage cell, an additional 121 ppm of soluble boron is required to maintain $k_{eff} \leq 0.95$. The soluble boron concentration required to maintain $k_{eff} \leq 0.95$ under normal operating conditions is 250 ppm, therefore, a total soluble boron concentration of 371 ppm is required to accommodate the limiting accident. This is well below the limit of 2000 ppm established in LCO 3.7.15.

The configuration of fuel assemblies in the cask storage area satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The k_{eff} of the spent fuel cask will always remain ≤ 0.95, assuming the spent fuel pit, including the cask storage area, is flooded with borated water and <1.0 with unborated water. The acceptable combination of initial enrichment and burnup are specified in Figure 3.7.16-1 for the Cask Storage Configuration.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the cask storage area of the spent fuel pit.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If the LCO is not met in MODE 5 or 6, LCO 3.0.3 would not be applicable. Moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, is independent of reactor operation. Therefore, inability to move the noncomplying fuel assembly to an acceptable storage location, is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the cask storage area is not in accordance with the acceptable combination of initial enrichment and burnup, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.16-1.

PROPOSED NEW SPECIFICATION BASES

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is within the acceptable burnup domain of Figure 3.7.16-1. This surveillance must be completed prior to placing any fuel assembly in the spent fuel cask.

REFERENCES

1. USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, June, 1987.

2. USNRC Spent Fuel Storage Facility Design Bases (for Comment) Proposed Revision 2, 1981.

3. ANS, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," ANSI/ANS-57.2-1983.

4. NET-239-02, "Criticality Analysis for Soluble Boron and Burnup Credit of the HI-STORM 100 Multi-Purpose Canister (MPC-32) in the Indian Point Unit 2 Spent Fuel Pool", Revision 2, Northeast Technology Corp.; Kingston, NY, 20 April 2006.

5. NRC, Letter to all Power Reactor Licensees from B.K. Grimes "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

ATTACHMENT 3 TO NL-06-020

CRITICALITY ANALYSIS FOR SOLUBLE BORON AND BURNUP CREDIT OF THE HI-STORM 100 MULTI-PURPOSE CANISTER (MPC-32) IN THE INDIAN POINT UNIT 2 SPENT FUEL POOL

Entergy Nuclear Operations, Inc.
Indian Point Unit No. 2
Docket No. 50-247

NET-239-02

CRITICALITY ANALYSIS FOR SOLUBLE BORON AND BURNUP CREDIT OF THE HI-STORM 100 MULTI-PURPOSE CANISTER (MPC-32) IN THE INDIAN POINT UNIT 2 SPENT FUEL POOL

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Review/Approval Record

Revision	Date	Prepared by:	Reviewed by:	Quality Assurance:
Ø	2/8/06	M. C. Harris	[illegible]	L. P. Marzani
1	3/7/06	M. C. Harris	[illegible]	L. P. Marzani
2	4/20/06	M. C. Harris	[illegible]	L. P. Marzani

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ABSTRACT

The analysis described herein was performed to support the licensing of an Interim Spent Fuel Storage Installation (ISFSI) at Entergy Nuclear Operations, Inc. (ENO)'s Indian Point Energy Center (IPEC) in Buchanan, New York. The ISFSI will utilize the Holtec International Hi-Storm 100 Cask System using the thirty-two assembly multi-purpose canister (MPC-32). This analysis supports a license amendment request (LAR) for a Technical Specification change for cask loading operations in the IPEC Unit 2 spent fuel pool and satisfies the criticality accident requirements of both 10CFR Part 50 for spent fuel pools and 10CFR Part 72 for ISFSIs.

NRC Regulatory Issue Summary (RIS) 2005-05[1] alerted licensees to regulatory inconsistencies which were identified during review of site specific ISFSI license applications. The inconsistencies are related to the criticality analysis methodologies which were utilized. RIS 2005-05 also states that the NRC staff determined that one potential resolution strategy to ensure compliance with 10CFR 50.68 is to perform new 10CFR Part 50 criticality analyses for fuel loaded into dry storage casks in accordance with previously accepted Part 50 conservatisms and assumptions.

A criticality analysis was performed using a methodology approved previously by the NRC for soluble boron and burnup credits in the IP2 spent fuel pool[2]. The analysis documented in this report was performed for a fully loaded, thirty-two assembly, multi-purpose canister (MPC-32) and transfer cask positioned in the cask pit in the southwest corner of the IP2 spent fuel pool. It was determined that under conditions of maximum reactivity, when loaded with fuel of the maximum allowable enrichment (1.8 w/o ^{235}U at zero burnup or up to 5.0 w/o ^{235}U with credit for burnup and IFBAs), that k_{eff} is less than 1.0 without credit for soluble boron during cask loading operations. In addition, it was determined that under the same assumptions, a soluble boron concentration of 250 ppm will maintain $k_{eff} \leq 0.95$ during the loading of fuel assemblies into the MPC-32, while in the spent fuel pool. Three (3) off-normal accident conditions (i.e., mis-loading a fresh fuel assembly into the MPC, placement of a fully loaded MPC-32 adjacent to the most reactive spent fuel modules or the accidental dropping of the maximum reactivity fuel assembly onto the fully loaded MPC-32) were evaluated. For the bounding case, an additional 121 ppm of soluble boron (a total soluble boron concentration of 371 ppm) is required to maintain $k_{eff} \leq 0.95$. 2

Boron dilution analyses previously performed[3] determined the dilution volumes required to dilute the spent fuel pool from 2000 ppm to 786 ppm soluble boron, the lower value being the concentration required to maintain $k_{eff}<0.95$. This latter concentration (786 ppm), also includes soluble boron credit for uncertainties related to burnup, and thus is higher than the concentration required (250 ppm) to reduce Δk_{eff} by 0.05 in order to accommodate the MPC under normal conditions and the concentration (371 ppm) required to accommodate the MCP under credible accident conditions. Administrative procedures previously incorporated[4] in the revised Plant Technical Specifications will prevent a dilution from occurring, which could potentially reduce the spent fuel pool boron concentration to a value 2 2

that would result in k_{eff} being greater than 0.95. For the present analysis, the soluble boron concentration was assumed to be only that required to reduce k_{eff} by 0.05, therefore the boron dilution analysis remains valid and it is conservative with respect to dilution volumes and times.

1.0 Introduction

In 1990, the Indian Point Energy Center (IPEC) Unit No. 2 spent fuel racks (SFRs) were replaced with new SFRs to increase the on-site storage capacity for spent fuel. Region 1 racks were designed to accommodate fresh fuel with enrichments up to 5.0 $^{w}/o$ ^{235}U. Region 2 racks accommodate much lower enrichment fresh fuel, and also accommodate higher enrichment fuel that has undergone burnup -- e.g., 1.764 $^{w}/o$ ^{235}U at zero burnup, or 5.0 $^{w}/o$ ^{235}U at 40,900 MWD/MTU[5]. The capacity in the pool was increased by decreasing the spacing between adjacent fuel assemblies. This decreased spacing was compensated for by using neutron absorbers between rack cells in order to maintain a sufficiently sub-critical configuration. In both the Region 1 and Region 2 IP2 SFRs, panels of Boraflex are used to control the reactivity of the fuel.

Since Boraflex is susceptible to in-service degradation, a RACKLIFE[6,7] model of the IPEC Unit 2 spent fuel pool was developed[8]. The analysis indicated that areas of moderate dissolution of the Boraflex panels had likely occurred. BADGER[9] tests performed initially in February 2000 (and again in July 2003) confirmed the predictions of the RACKLIFE computer model[10,11]. Criticality analyses were repeated in 2001 to take credit for soluble boron and burnup in order to offset the reactivity effects of degraded Boraflex as indicated by the BADGER test results[2]. The criticality analysis followed a methodology previously approved by the NRC[12]. The analysis showed that sufficient sub-critical margin could be maintained through 2006, at which time the IPEC Unit 2 spent fuel racks would lose full core offload capability.

In order to maintain full core offload capability, ENO has chosen to install an ISFSI at the IPEC, which will utilize the Hi-Storm 100 Storage System. This report describes the criticality analysis of the IPEC Unit 2 spent fuel pool with a fully loaded multi-purpose canister (MPC) containing thirty-two fuel assemblies of maximum reactivity. This analysis takes credit for soluble boron in the spent fuel pool water and parallels the analyses performed previously for the SFRs[2].

The principal design criteria applied to the IPEC Unit 2 SFRs with degraded Boraflex is $k_{eff} < 1.0$ with no soluble boron (including all biases, tolerances and uncertainties), and $k_{eff} \leq 0.95$ with credit for soluble boron. The maximum soluble boron credit required by the current design basis in all SFR sub-regions is 786 ppm, in order to maintain $k_{eff} \leq 0.95$ for normal conditions and 1495 ppm for accident conditions. The previous analyses iteratively determined the required soluble boron concentrations. For this analysis of a fully loaded MPC in the spent fuel pool cask pit, it is demonstrated that the previously established concentrations are sufficient to maintain $k_{eff} \leq 0.95$.

2.0 Descriptions of the Indian Point 2 Fuel Pool, Storage Racks, and Fuel

2.1 Pool Configuration

The layout of the rack modules in the IP2 fuel pool is shown in Figure 2-1. The Region 1 racks consist of three spent fuel storage rack modules (269 total storage locations) of the flux trap design. The Region 2 racks consist of nine spent fuel storage rack modules (1,105 locations) of the egg-crate design. The total capacity of the pool is 1,374 storage cells.

2.2 Cask Pit and MPC Design Features

During loading operations, the MPC-32 multipurpose canister is placed inside a Hi-Trac transfer cask centered within the cask pit in the southwest corner of the spent fuel pool. The effective dimension of the cask pit (from the pool wall to storage racks) is 103 inches in the north-south direction and 94.375 inches in the east-west direction. The Hi-Trac transfer cask contains a water jacket for shielding with an outside diameter of 88.75 inches. The outer diameter of the MPC shell is 68.375 inches.

The MPC storage basket contains 32 stainless steel square storage cells made of 9/32 inch thick walls on a 9.218 inch cell pitch as shown in Figure 2-2. The basket contains 16 inner storage cells with two poison panels affixed to adjacent walls via stainless steel sheathing, creating an inner dimension of 8.79 inches. Fourteen of the sixteen outer storage cells have a single poison panel. These cells have an inside dimension of 8.79 inches between the cell wall and the poison panel. The cells have an 8.97 inch inside dimension between the opposing non-poison cell walls. Two storage locations contain no poison panels.

The poison material may be one of two neutron absorbers: BORAL or Metamic. The minimum areal density for BORAL is 0.0372 grams $^{10}B/cm^2$. The minimum areal density for Metamic is 0.0310 grams $^{10}B/cm^2$.[13]

2.3 Region 2 Rack Cell Design Features

The details of this egg-crate rack design are illustrated in Figure 2-3. The basic storage cell for the Region 2 racks consists of three primary elements: 1) the fuel boxes, 2) the Boraflex panels, and 3) cover plates to retain the Boraflex. The boxes are welded corner to opposite corner to form a grid of alternating cells with fuel boxes, between which additional cells are formed. (An assembly in a box faces the box walls; an assembly between boxes faces the cover plates of the four cells that surround it).

The fuel boxes are formed from 0.075" thick sheets of Type 304 stainless steel. They are 169" long and are welded to the rack module base plate. The Boraflex panels are

nominally 0.082" thick, 7.5" wide, and 150" long in the Region 2 racks. The neutron absorber boron-10 nominal areal density is 0.0260 g/cm^2. The cover plates are 0.035" thick Type 304 stainless steel and cover the entire length and width of the Boraflex with additional material on all sides bent to form a cavity 0.092" thick at the face of the Boraflex. The assembly-to-assembly pitch is 9.04" and the box inside dimension is 8.8", while the cover plate to cover plate dimension is 8.876". The Region 2 racks are diagonally symmetric.

2.4 Westinghouse 15x15 Fuel Assemblies

Subsequent analyses described in this report are based specifically on a conservative model of the Westinghouse 15x15 assembly design, as detailed in Table 2-1. The IP2 SFRs contain HIPAR design assemblies (with Inconel spacer grids, stainless steel guide tubes, and Zircaloy clad), LOPAR design assemblies (with Inconel spacer grids and Zircaloy guide tubes and clad), OFA design assemblies (with Zircaloy spacer grids, guide tubes, and clad), Vantage+ (with Zircaloy spacer grids, ZIRLO guide tubes, and ZIRLO clad), Performance+ and 15x15 Upgraded Fuel Assemblies. (It is noted that the top and bottom spacer grids of all designs are Inconel.) Vantage+, Performance+ and 15x15 Upgraded design assemblies are all essentially similar except in the number and placement of grid spacers and the length of the low enriched annular blankets at the ends of the fuel rods, neither of which are taken credit for in this analysis. The OFA, LOPAR, and HIPAR assemblies are uniformly enriched.

The Vantage+, Performance+ and 15x15 Upgraded design fuel pellet diameters are identical to the OFA, LOPAR, and HIPAR rods, but the fuel density is slightly higher, making these the highest reactivity assemblies to be stored in the racks. In addition, in the Vantage+, Performance+ and 15x15 Upgraded designs, the cladding material, ZIRLO, has a slightly higher absorption cross section than Zircaloy. Sensitivity analyses of all assembly types has shown that when the effects of leakage are incorporated, as in the current analyses, the differences in calculated values of k_{eff} for the various types of fuel assemblies are negligible. The primary geometric difference between the assembly types is the radial dimension of the instrument and guide tubes. The OFA design has a marginally smaller guide tube, which displaces a lesser volume of moderator and is thus slightly more reactive. Thus, a maximum reactivity OFA bundle (with its smaller guide tube and Zircaloy clad) with a maximum fuel density appropriate for the Vantage+ based fuel types was conservatively used for the analyses.

Table 2-1: Westinghouse 15 x15 Fuel Assembly Description

FUEL RODS	
Cladding Material	Zircaloy
Cladding Tube OD	0.422 in
Cladding tube wall thickness	0.0243 in
Pellet material	Sintered UO_2
Pellet OD	0.3659 in
Pellet density, % theoretical	95.7%
Pellet-to-clad diametral gap	0.0075 in
Total Fuel Rod Length	144.0 in
Water Rod OD	0.532 in
Water Rod ID	0.498 in
FUEL BUNDLES	
Number of Fuel Rods (# of water rods)	204 (21)
Rod array	15 x 15
Rod-to-rod pitch	0.563 in
Bundle dimensions	8.445 in x 8.445 in
Maximum enrichment, $^{w}/o$ ^{235}U*	5.0

*Maximum enrichment without credit for IFBAs is 4.50 w/o ^{235}U

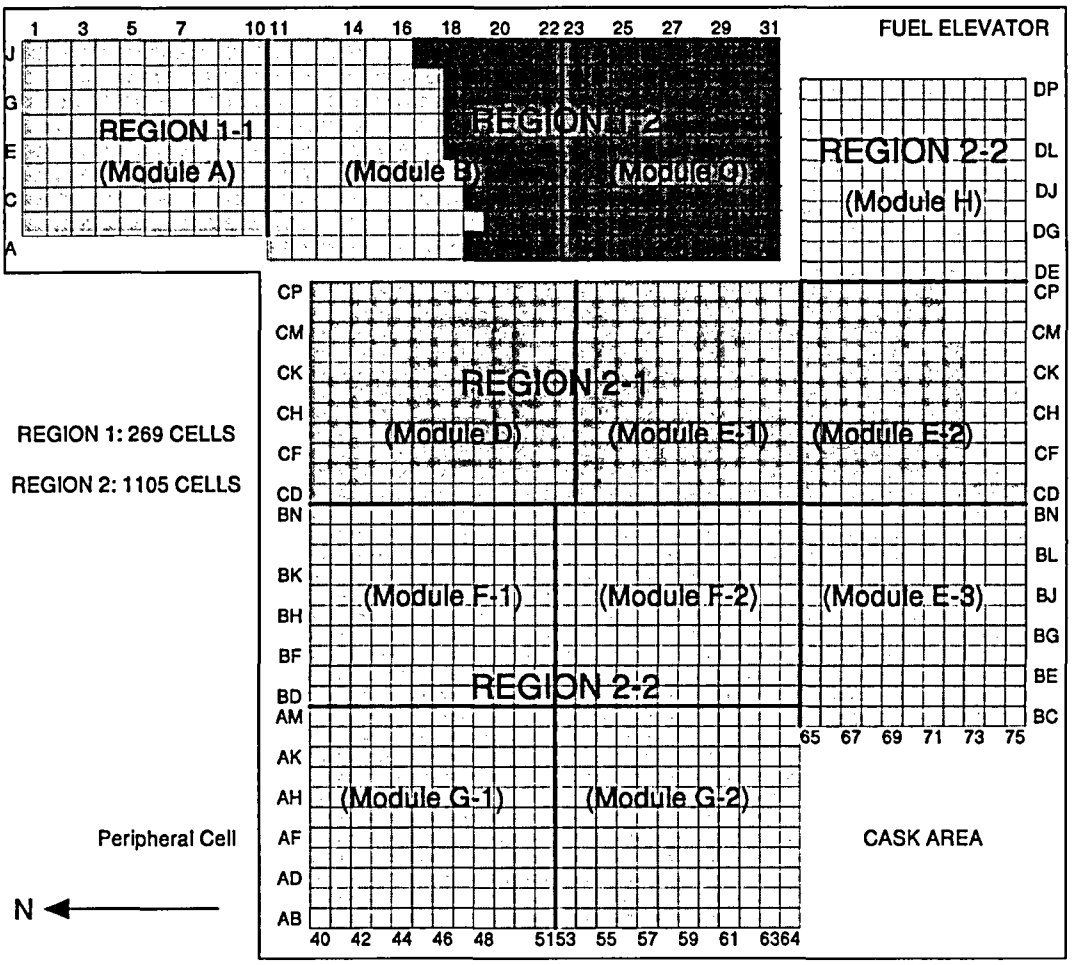


Figure 2-1: Indian Point 2 Spent Fuel Pool Layout with Module IDs

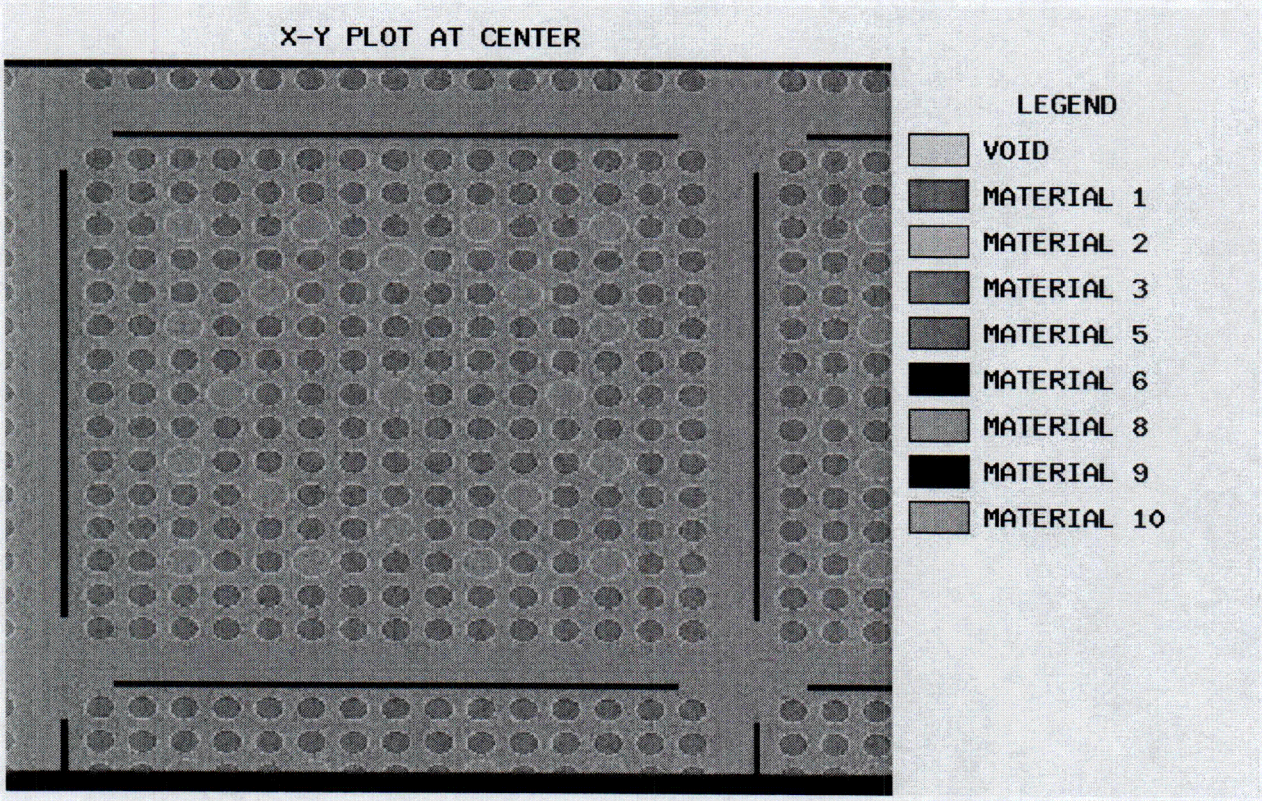


Figure 2-2: MPC-32 Storage Cell

Key

Material Number	Material
1	UO_2 (1.8 w/o)
2	Zircaloy
3	Unborated Water
5	Stainless Steel
6	Boraflex
8	Lead
9	Boron Carbide
10	Aluminum

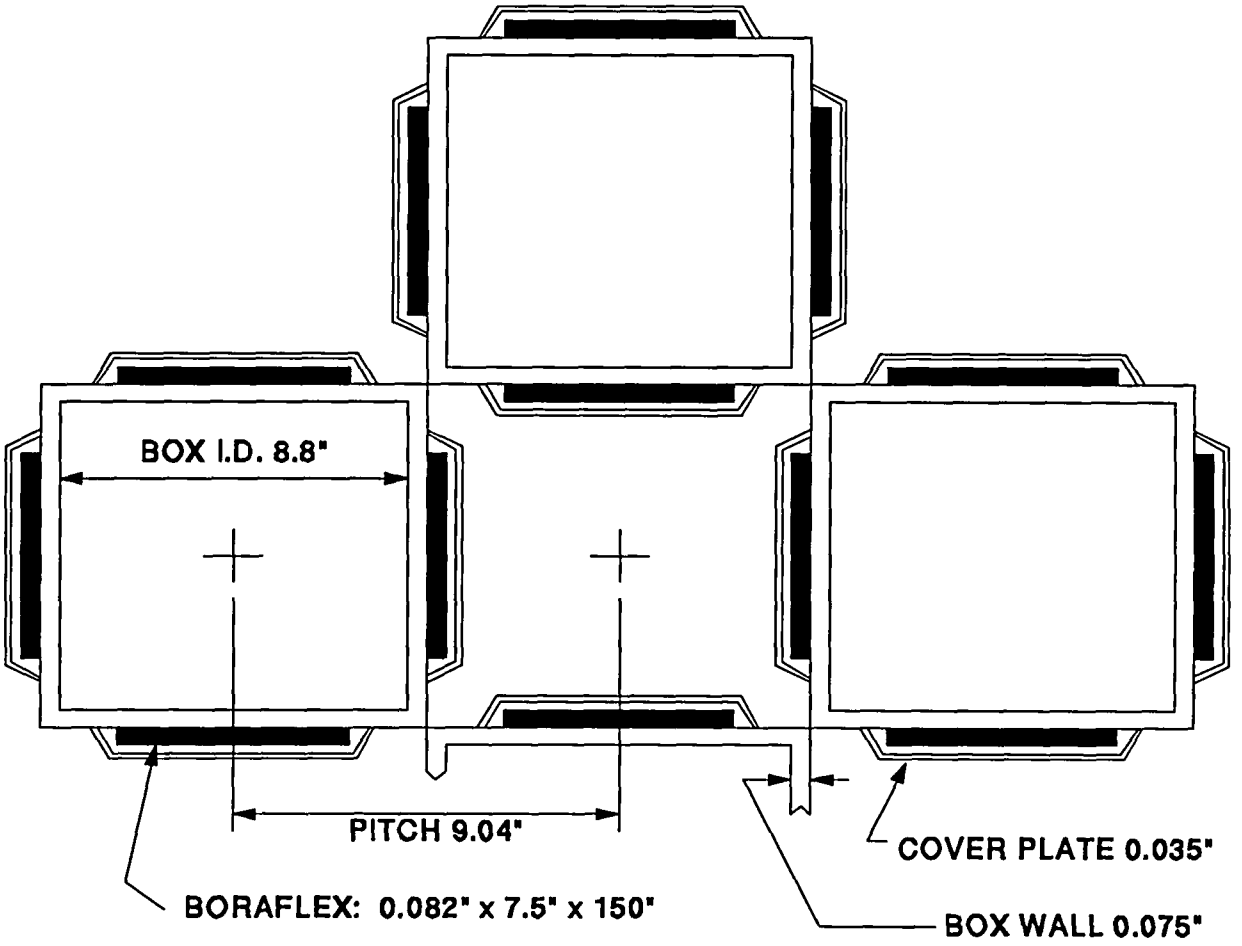


Figure 2-3: Indian Point 2 Region 2 Storage Cells

3.0 Methods and Computer Codes

3.1 Soluble Boron Credits

The methodology used to take credit for soluble boron and burnup parallels the analyses performed previously for the Indian Point 2 spent fuel pool [2,12]. While the basic approach is the same, some differences from the referenced method occur, in particular:

- Computer codes used in the analysis (i.e., KENO V.a, ORIGEN)
- Finite pool modeling (including Region 2-2 and adjacent Cask Pit) rather than an infinite array of cells.
- Additional tolerances included in statistical combinations (i.e., pellet OD, cladding OD, cladding ID, guide tube OD, guide tube ID)
- Soluble boron concentrations were assumed to be at the limiting values as determined previously [2].

The basic approach for taking credit for soluble boron and burnup in the loaded MPC is as follows:

1) Determine the maximum enrichment at zero burnup, such that at a 95% probability with 95% confidence, k_{eff} (including tolerances and uncertainties) is less than 1.0 for unirradiated fuel.

2) Employ the technique of reactivity equivalencing to determine, through depletion analyses, the minimum assembly burnup required (as a function of initial enrichment) for each sub-region of the SFRs and MPC.

3) Determine the limiting maximum required soluble boron concentration for normal conditions to reduce k_{eff} by 0.05. Verify that at the enrichment determined in Step 1 above and at a 95% probability with 95% confidence, k_{eff} (including tolerances and uncertainties) is less than 0.95 for unirradiated fuel. Since the IPEC Unit 2 SFRs already take credit for soluble boron, the limiting value for normal conditions determined for an infinite array of cells was assumed for the analysis of a fully loaded MPC in finite geometry[2,3].

3.2 Computer Codes

This analysis was performed primarily with the stochastic Monte Carlo code KENO V.a[14]. Independent confirmatory calculations were completed with the monte-carlo transport code MCNP[15]. The point depletion ORIGEN code, also contained in the SCALE 5 code package, was used to generate burnup dependent isotopics as input into Keno V.a for analyzing axial reactivity effects. Depletion uncertainties were calculated using the deterministic code CASMO-4[16].

KENO V.a is a module in SCALE 5, a collection of computer codes and cross section libraries used to perform criticality safety analyses for licensing evaluations. KENO solves the three-dimensional Boltzmann transport equation for neutron-multiplying systems. The collection also contains BONAMI-S to prepare problem specific master cross section libraries and to make resonance self-shielding corrections for nuclides with Bondarenko data. NITAWL-II is used to prepare a working cross section library with corrections for resonance self-shielding using the Nordheim integral treatment. These modules are invoked automatically by using the CSAS25 analysis sequence in SCALE 5.

CASMO-4 is a two dimensional multigroup transport theory code for fuel assembly burnup analysis in-core or in typical fuel storage racks. CASMO is a cell code in which infinitely repeating arrays of fuel assemblies and/or fuel racks are modeled.

These codes have been verified and validated for use in spent fuel rack design evaluations by using them to model a number of critical experiments[17-20]. The results of this validation and verification are included in this report as Appendix A[21]. The calculated k_{eff} was compared to the critical condition (k_{eff} = 1.0) to determine the bias in the calculated values.

In all SCALE/KENO calculations, the 238 energy group ENDF/B-V criticality safety cross-section library[22] was used. The resulting bias in the SCALE codes was calculated to be -0.0078± 0.0036. In all CASMO calculations, the CASMO standard 70 energy group cross-section library was used. The resulting bias in the CASMO code was calculated to be -0.0103 ± 0.0020. In all MCNP calculations, the continuous energy cross-section libraries, based on ENDFB-VI, were used. The resulting bias in the MCNP5 code was calculated to be –0.0057 ± 0.0051.

For all KENO and MCNP calculations, a one-sided 95% probability / 95% confidence statistical tolerance factor is applied to the computed eigenvalue. In all KENO runs, typically 3000 generations (after skipping between 50 and 100 generations for source distribution convergence) with typically 2000 neutrons per generation were simulated, for a total of 6 million neutrons tracked. This typically resulted in statistical uncertainties in k_{eff} of $\sigma < 0.0002$ (one standard deviation) and a 95/95 statistical tolerance factor $\kappa \approx 1.7$[23].

4.0 Assumptions, Biases, and Uncertainties for Criticality Analysis

4.1 Reference Analysis

During cask loading, the MPC-32 resides in a Hi Trac transfer cask in the cask loading area in the southwest corner of the pool. (See Figure 2-1). The reference Keno V.a model encompasses the entire Region 2-2 spent fuel storage racks and cask pit and is an explicit geometric representation of the Region 2-2 spent fuel racks. The model utilizes concrete albedos on the north, west and south boundaries of the spent fuel pool and a water albedo on the east periphery. Consistent with the current IP2 Plant Technical Specifications, the Boraflex in the Region 2-2 racks is assumed to be degraded. The individual Boraflex panels are assumed to have thinned to 70% of the minimum certified as-manufactured thickness. Burnup credit has been assumed in the current analysis of record, such that the equivalent maximum enrichment of an unirradiated assembly permitted in Region 2-2 is 1.80 w/o U-235[2].

The reference model consists of a fully loaded MPC-32 with fuel assemblies of the maximum permissible enrichment. As shown in Figure 4-1, the MPC-32 is positioned inside the Hi-Trac transfer cask, which contains lead for gamma shielding and a water-filled jacket for neutron shielding. In the reference case analysis, the MPC is assumed to be centered geometrically in the cask pit. Tables 4-1 contains the best estimate k_{eff} for the reference calculation, as modeled with BORAL neutron absorber panels. k_{eff} values are given both for the reference KENO V.a calculation, as well as MCNP5, which serves as an independent check calculation.

In accordance with standard practice[12,24], the reactivity effects of the following tolerances and uncertainties were analyzed:

- Pellet Diameter
- Cladding Inner Diameter
- Cladding Outer Diameter
- Minimum Clad Thickness
- Guide Tube Inner Diameter
- Guide Tube Outer Diameter
- Cell Inner Dimension Tolerance
- Cell Wall Thickness Tolerance

- Cell Pitch Tolerance
- ^{235}U Enrichment Tolerance
- UO_2 Density Tolerance
- Asymmetric Assembly Position Tolerance
- Fuel Pellet Dishing
- Methodology Bias Uncertainty (at 95/95)
- Calculation Uncertainty (at 95/95)

Table 4-2 contains the manufacturing tolerances, as determined via manufacturing specifications, as-built drawings, or benchmark calculations. Column 1 lists the specific tolerance, while Column 2 lists the actual value. Because the neutron absorber is modeled at its minimum certified dimensions and ^{10}B loading, no sensitivity analysis with respect to neutron absorber dimensions on loading are necessary.

In addition, the following biases were also accounted for:

- Calculation Methodology Bias
- Reactivity Equivalencing Bias
- Discrete Absorber Particle Self-Shielding Bias

The first two biases listed are implicitly incorporated into the results reported subsequently in Section 5.0. The calculation methodology bias is based on the verification and validation of the computer codes used, as discussed in Section 3.2. The reactivity equivalencing bias accounts for potential deficiencies in the methodology of equivalencing the reactivity of depleted fuel to that of a fresh fuel assembly at a lower enrichment[25].

The discrete absorber particle self-shielding bias accounts for the fact that the neutron absorbers are made from discrete boron carbide particles and thus are not a homogeneous distribution of absorber values. The self-shielding correction factor was determined, based on the boron carbide particle size distribution of a large particle neutron absorber (BORAL), thus it is conservative when applied to a fine particle absorber such as Metamic.

4.2 Uncertainties Introduced by Depletion Analyses

4.2.1 Assembly Burnup

Benchmarks were performed to assess the effects of burnup dependent cross-sections on the associated uncertainty in the reactivity of the fuel storage racks based on Post Irradiation Examinations (PIEs) of fuel rods taken from various depleted PWR fuel assemblies[24]. Measured isotopics for the principal isotopes with respect to reactivity, specifically, ^{235}U, ^{239}Pu and ^{241}Pu, as well as ^{234}U, ^{236}U, ^{238}U, ^{238}Pu, ^{240}Pu and ^{242}Pu were compared with predicted values. The depletion cycle that most closely models IP2 operation was selected.

To assess the reactivity effects introduced by uncertainties in burnup dependent isotopics in a 15x15 fuel assembly, a single rod (G9) of assembly D01 irradiated through Cycles 2, 3 and 4 at Turkey Point 3 was modeled with CASMO-4 for benchmarking purposes. The CASMO-4 model was depleted to the actual rod burnup (30.72 GWD/MTU) using the operating history reported[26]. After depletion, an assembly consisting of only G9 rods with predicted isotopics was placed in a fuel rack and the k_{eff} calculated. The assembly and racks were again modeled with isotopics, as measured in the PIEs and again the k_{eff} was calculated. The associated reactivity due to uncertainty in the predicted versus measured isotopics was +0.00330 Δk at 30.72 GWD/MTU. It is generally accepted that reactivity varies linearly as a function of burnup and subsequently, the uncertainty as well. This extrapolates to an uncertainty of +0.00660 Δk at 61.44 GWD/MTU. As a conservative upper bound for this analysis, it will be assumed that the uncertainty due to depletion dependent isotopics is +0.007Δk at 60 GWD/MTU. This is nearly twice the minimum burnup for a rod with an initial enrichment of 5.0 w/o, as determined in Section 5.3, and is therefore conservative.

4.2.2 Axial Burnup Effect on Reactivity

In addition to the uncertainty in reactivity resulting from depletion dependent isotopics, there exists the possibility of a reactivity increase due to non-uniform axial depletion of the fuel assembly. In general, most cell depletion codes (i.e., CASMO, CPM, etc.) are 2-D codes that utilize an implicit uniform axial power shape. Certain conditions can occur in the reactor (e.g., control rod insertion, coolant void, etc.) that can affect the isotopic burnup and depletion, causing a higher reactivity than that associated with a uniform burnup distribution.

Analysis of the available data indicated that in IP2 Cycles 14 and 15, there exists a single assembly (R08) that was located in the central core location (H-8), where a rod control cluster assembly (RCCA) from Control Bank D is normally inserted to the "bite-position". Normally, assemblies in this position will be shuffled to a non-rodded location in the subsequent cycle, but R-08 appears to be a unique case. Operation with Control Bank D at the bite position can actually cause a flux depression at the top of the fuel and result in a lower burnup relative to similar assemblies that are in non-rodded

locations. Projected region burnups for 24 axial regions for Cycles 14 and 15 were evaluated.

The SCALE5 depletion module SAS2H was utilized to determine the depletion dependent isotopics for each of the 24 axial regions for assembly R08 at the end of Cycles 14 and 15. The generated isotopics were then input into a discrete 24 axial region KENO V.a model of the racks and the k_{eff} was calculated. A discrete axial model was created for the fully loaded MPC.

As a reference comparison, similar uniform axial burnup models were created for each SFR sub-region. These models assume that all regions are at the same assembly average burnup. Again, the SAS2H depletion sequence was employed to produce depletion dependent isotopics for the reference model. The difference between the k_{eff} of the 24-axial region explicit model and the uniform axial model were calculated. The resulting reactivity effect of a fully loaded MPC with assemblies having the largest deviation in non-uniform axial burnup is +0.00100Δk at a 95% probability with 95% confidence.

4.2.3 Removal of Burnable Absorbers

Both the current Vantage+ based fuel types (which include the Perfomance+ and 15x15 Upgraded design) and earlier fuel types utilize removable burnable poison assemblies. These include the Wet Annular Burnable Absorbers (WABAs), which are normally removed after one cycle of operation. Since the WABAs tend to harden the neutron spectrum, an assembly containing WABAs may not achieve as high a burnup as an assembly without WABAs.

The reactivity effect of fuel depletion with WABAs was also assessed. This condition was analyzed by depleting an assembly with and without the maximum number of WABAs contained in a 15 x 15 assembly. Subsequently, the reactivities (in a cold rack condition) were compared as a function of burnup. The assembly with WABAs present during irradiation was depleted and the WABAs removed prior to placement in the rack in the cold condition. The k_{eff} of this configuration was compared to the same assembly that never contained WABAs and the Δk computed as a function of burnup. The maximum difference was +0.00951 Δk.

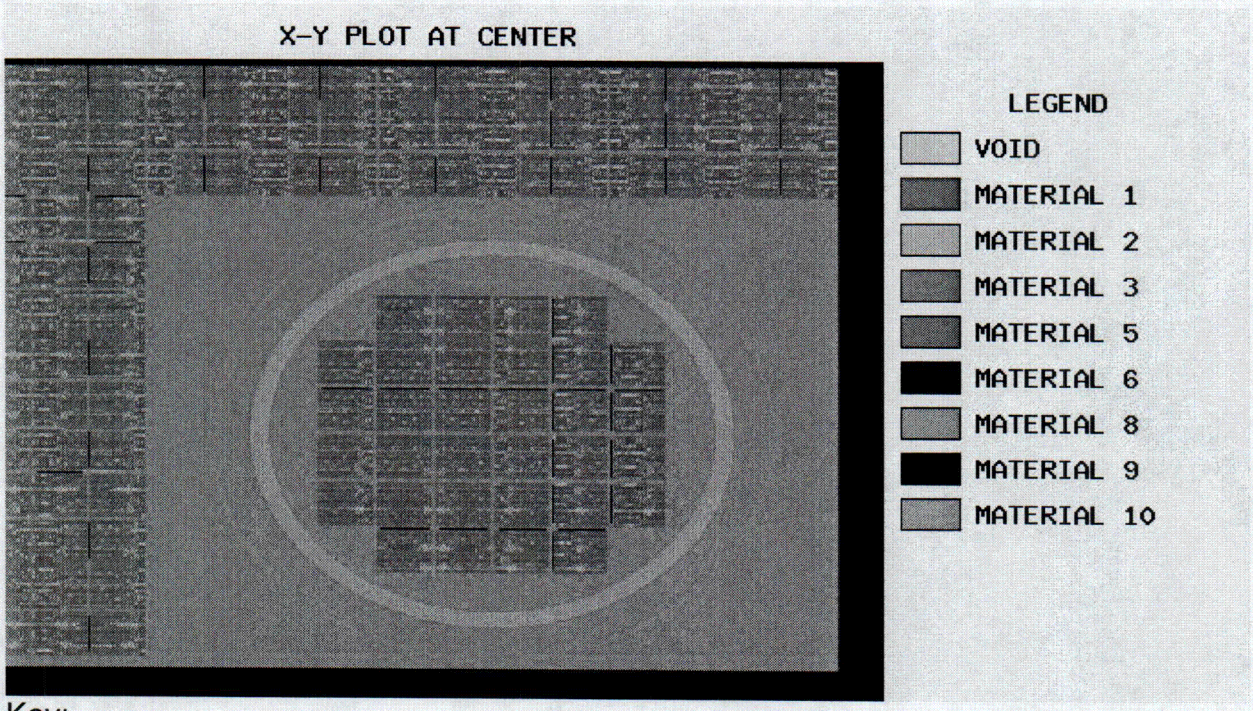
Table 4-1: Comparison of k_{eff} Reference Case MPC-32 Utilizing BORAL Absorbers with No Soluble Boron

Reference Case	Keno V.a	MCNP4B
k_{eff}	0.97031	0.96656

Table 4-2: Fuel Assembly and Cell Tolerances for the MPC-32 Multipurpose Canister

Tolerance	Value(inches)
Pellet Diameter*	±0.0005
Cladding Inside Diameter*	±0.0015
Cladding Outside Diameter*	±0.0015
Clad Minimum Thickness*	0.0225
Cell ID	±0.06
Cell Wall Thickness	±0.007
Cell Pitch	±0.06
Guide Tupe I.D.	±0.002
Guide Tube O.D.	±0.002
Enrichment ($^{w}/o$ ^{235}U)	±0.05
UO_2 Density	±2%
Fuel Pellet Dishing	0%-2% (0%-1% each end)
Asymmetric Fuel Position	Offset (relative to Centered)

*Taken from Reference 27



Key:

Material Number	Material
1	UO_2 (1.8 w/o)
2	Zircaloy
3	Unborated Water
5	Stainless Steel
6	Boraflex
8	Lead
9	Boron Carbide
10	Aluminum

Figure 4-1: Two Dimensional Cross-Sectional Plot of Loaded MPC-32 in Cask Pit

5.0 Results of the Criticality Analysis

5.1 Reference Model, Including Tolerances and Uncertainties

The reference case model of a fully loaded MPC utilizing METAMIC neutron absorber panels in the IPEC Unit 2 cask pit results in a k_{eff} of 0.97108. This model assumes a nominal pool temperature of 68°F (corresponding to room temperature at which cross-section data is evaluated) with water at full density and no soluble boron. Table 5-1 presents the associated reactivity effects for each of the manufacturing tolerances. The statistical combination of uncertainties and tolerances adds an additional 0.01794Δk to the reference k_{eff}, resulting in a 95% probability at 95% confidence (95/95) upper statistical limit on k_{eff} of 0.99414. This is less than 1.0 without credit for soluble boron. 1

The reference fully loaded MPC utilizing BORAL neutron absorber panels in the IPEC Unit 2 cask pit results in a k_{eff} of 0.97031. Again, the reference model assumes the rack is at a nominal pool temperature of 68°F with water at full density and no soluble boron. Table 5-2 presents the associated reactivity effects for each of the manufacturing tolerances. The statistical combination of uncertainties and tolerances adds 0.01873Δk to the reference k_{eff}, resulting in a 95% probability at 95% confidence (95/95) upper statistical limit on k_{eff} of 0.99413. This is less than 1.0 without credit for soluble boron.

5.2 Soluble Boron Credit

In order to assure that k_{eff} remains below 0.95 with soluble boron, credit for limited soluble boron (250 ppm) is used. This lowers the 95/95 k_{eff} with soluble boron to $k_{eff} \leq 0.95$. Columns "2" of Tables 5-1 and 5-2 summarize the reference model k_{eff} and reactivity effects due to tolerances and uncertainties for the soluble boron concentration of 250 ppm boron.

5.3 Burnup Credit

Through reactivity equivalencing, the minimum burnup required that results in the same k_{eff} as that of a fresh bundle at 1.80 $^{w}/o$ was determined via depletion analysis with the CASMO-4 code. Since fuel in Region 2-2 may reside there for long periods of time, reactivity credit for decay of ^{241}Pu is taken into account. Figure 5-1 shows the minimum burnup as a function of initial enrichment for assemblies discharged into Region 2-2 as a function of fuel cooling time. As described previously, these minimum burnup curves have been adjusted by 4% to account for the uncertainty in calculated burnup. Table 5-3 contains the equations generated from best fit of the minimum burnup curves in Figure 5-1.

As described in Section 4.2.2, the reactivity effect due to non-uniform axial isotopic depletion was analyzed. The maximum reactivity effect due to reduced depletion in the upper axial nodes results in a 95/95 increase in Δk of +0.0010.

Table 5-1: Reactivity Changes Associated with Tolerances and Biases for Fully Loaded MPC-32 with Metamic Neutron Poison Panels

Item	No Boron	With Boron (250 ppm)
Reference (Calculation and Reactivity Equivalence Bias Corrected)		
k_{eff}	**0.97108**	**0.92243**
Tolerances and Uncertainties		
Pellet OD	0.00032	0.00021
Cladding ID+	0.00105	0.00190
Cladding OD	0.00130	0.00127
Guide Tube ID	0.00034	0.00041
Guide Tube OD	0.00021	0.00061
Cell ID	0.00032	0.00038
Cell Wall Thickness	0.00005	0.00010
Cell Pitch	0.00012	0.00045
Enrichment ($^{w}/o$ ^{235}U)	0.00887	0.00906
UO_2 Density	0.00300	0.00383
Asymmetric Position*	0.00000	0.00000
Dishing**	0.00000	0.00000
Methodology	0.00951	0.00951
Calculation	0.00027	0.00025
Depletion	0.00700	0.00700
WABA	0.00951	0.00951
Axial burnup	0.00100	0.00100
Total (Statistical Combination)	**0.01794**	**0.01827**
Biases		
Self Shielding	0.00312	0.00312
Reactivity Equivalencing	0.00200	0.00200
Upper Statistical Tolerance Limit (95/95)		
k_{eff}	**0.99414**	**0.94582**

+Worst case cladding ID tolerance or minimum clad thickness
* Relative to centered
**Pellets assumed undished.

Table 5-2: Reactivity Changes Associated with Tolerances and Biases for Fully Loaded MPC-32 with BORAL Neutron Poison Panels

Item	No Boron	With Boron (250 ppm)
Reference (Calculation and Reactivity Equivalence Bias Corrected)		
k_{eff}	**0.97031**	**0.92238**
Tolerances and Uncertainties		
Pellet OD	0.00095	0.00083
Cladding ID+	0.00235	0.00162
Cladding OD	0.00233	0.00149
Guide Tube ID	0.00134	0.00071
Guide Tube OD	0.00058	0.00029
Cell ID	0.00075	0.00066
Cell Wall Thickness	0.00010	0.00010
Cell Pitch	0.00059	0.00029
Enrichment ($^{w}/o$ ^{235}U)	0.00945	0.00922
UO_2 Density	0.00399	0.00360
Asymmetric Position*	0.00000	0.00000
Dishing**	0.00000	0.00000
Methodology	0.00951	0.00951
Calculation	0.00027	0.00025
Depletion	0.00700	0.00700
WABA	0.00951	0.00951
Axial	0.00100	0.00100
Total (Statistical Combination)	**0.01873**	**0.01832**
Biases		
Upper Statistical Tolerance Limit (95/95)		
Self Shielding	0.00312	0.00312
Reactivity Equivalencing	0.00200	0.00200
k_{eff}	**0.99416**	**0.94582**

+Worst case cladding ID tolerance or minimum clad thickness
* Relative to centered
**Pellets assumed undished.

Table 5-3: Minimum Burnup versus Enrichment Curves for Various Cooling Times

Cooling Time (Years)	Equation
0	$-21.9539 + 36.6097 \cdot ln(E)$
5	$-19.7170 + 33.4991 \cdot ln(E)$
10	$-18.6633 + 31.6203 \cdot ln(E)$
15	$-17.9294 + 30.5329 \cdot ln(E)$
20	$-17.4897 + 29.8016 \cdot ln(E)$

E = Initial Enrichment, w/o U-235

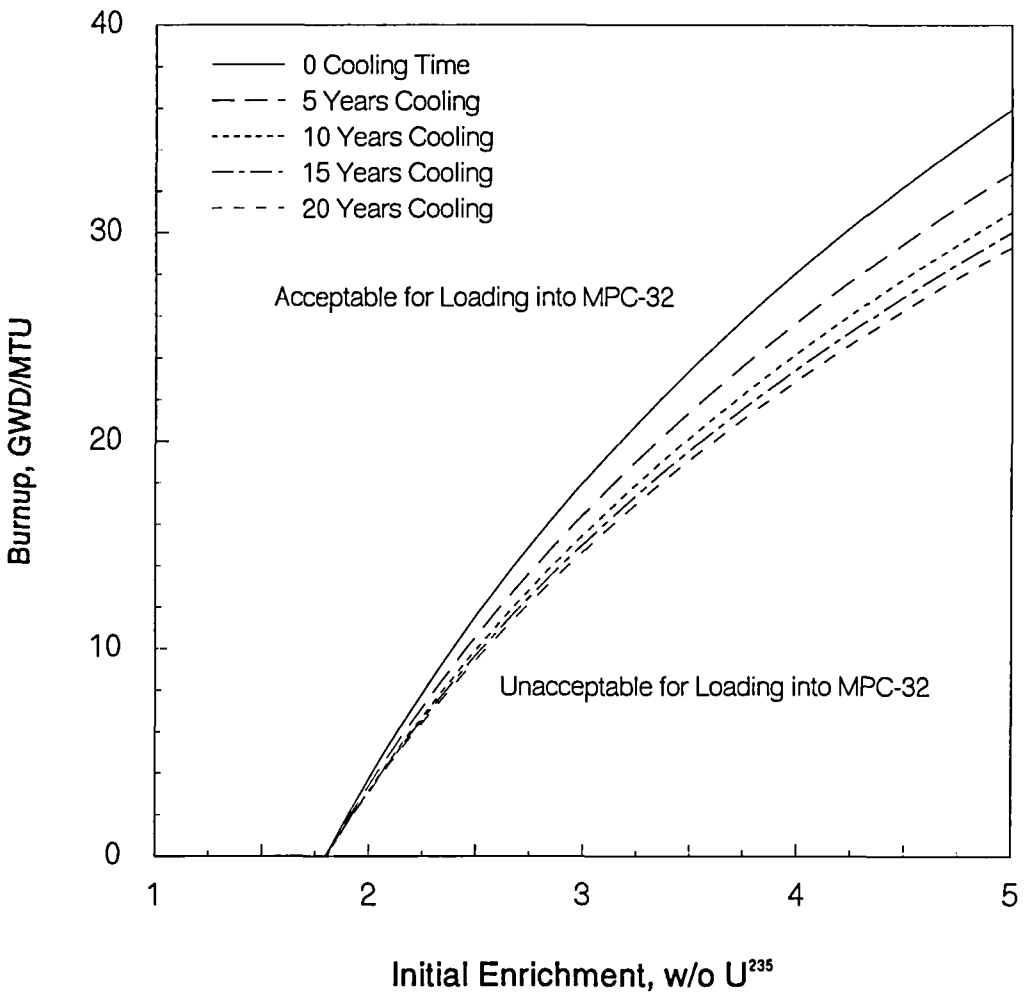


Figure 5-1: Minimum Assembly Burnup for a Fully Loaded MPC-32 in the Indian Point 2 Cask Pit

6.0 Accident Analysis

In addition to normal operating conditions, the occurrence of postulated abnormal occurrences have been analyzed per the requirements of ANSI/ANS-57.2-1983[28], Part 6.4.2.1.3. This analysis considered the following five categories of abnormal occurrences:

- MPC Positioned Alongside Spent Fuel Racks
- Fuel Assembly Dropped on Top of MPC or Dropped in Cask Pit Alongside Rack
- Abnormal Heat Load
- Seismic Event 1
- Fresh Fuel Assembly Mis-loaded into the MPC 2

Soluble boron credit is taken for the accident condition that results in the worst condition in terms of increased reactivity.

6.1 Cask Positioned Alongside the Region 2-2 Fuel Racks

In addition to the reference case with a fully loaded MPC-32 centered in the cask pit during normal loading operations, the off-normal case, where the Hi Trac cask containing the fully loaded MPC-32 is positioned adjacent to Modules G-2 and E-3, as shown in Figure 6-1, was also considered. The case without soluble boron was not considered, as this would constitute a second abnormal occurrence. For the off-normal case with soluble boron, the maximum k_{eff}, at the 95 percent probability/95 percent confidence level, was 0.94657 for the MPC with METAMIC as the neutron absorber.

6.2 Fuel Assembly Dropped on Top of a Fully Loaded MPC

The reactivity of a fresh fuel assembly dropped onto the top of the Hi-Trac cask containing a fully loaded MPC-32 was also analyzed. The dropped assembly is assumed to come to rest on the top of the Hi-Trac cask in a horizontal position as shown in Figure 6-2. This condition was evaluated with the cask centered in the cask pit. In addition, the length of a fuel assembly exceeds the N-S or E-W dimensions of the cask pit. It was therefore assumed that part of the assembly extends over and into the Region 2-2 fuel racks. 2

Again, the case without soluble boron was not considered, as this would constitute a second abnormal occurrence. For the off-normal case with soluble boron, the maximum k_{eff}, at the 95 percent probability/95 percent confidence level, was 0.94679 for the MPC with METAMIC as the neutron absorber.

6.3 Fuel Assembly Dropped In the Cask Pit

The reactivity of a fresh fuel assembly dropped in the cask pit containing a fully loaded 2

MPC-32 was analyzed. Various assembly positions were analyzed to determine the maximum reactivity configuration. The dropped assembly in the northeast corner of the cask pit, next to two adjacent SFRs, as shown in Figure 6-3, was determined to be the most reactive.

Again, the case without soluble boron was not considered, as this would constitute a second abnormal occurrence. For the off-normal case with soluble boron, the maximum k_{eff}, at the 95 percent probability/95 percent confidence level is 0.94931 for the MPC with METAMIC as the neutron absorber.

6.4 Abnormal Heat Load

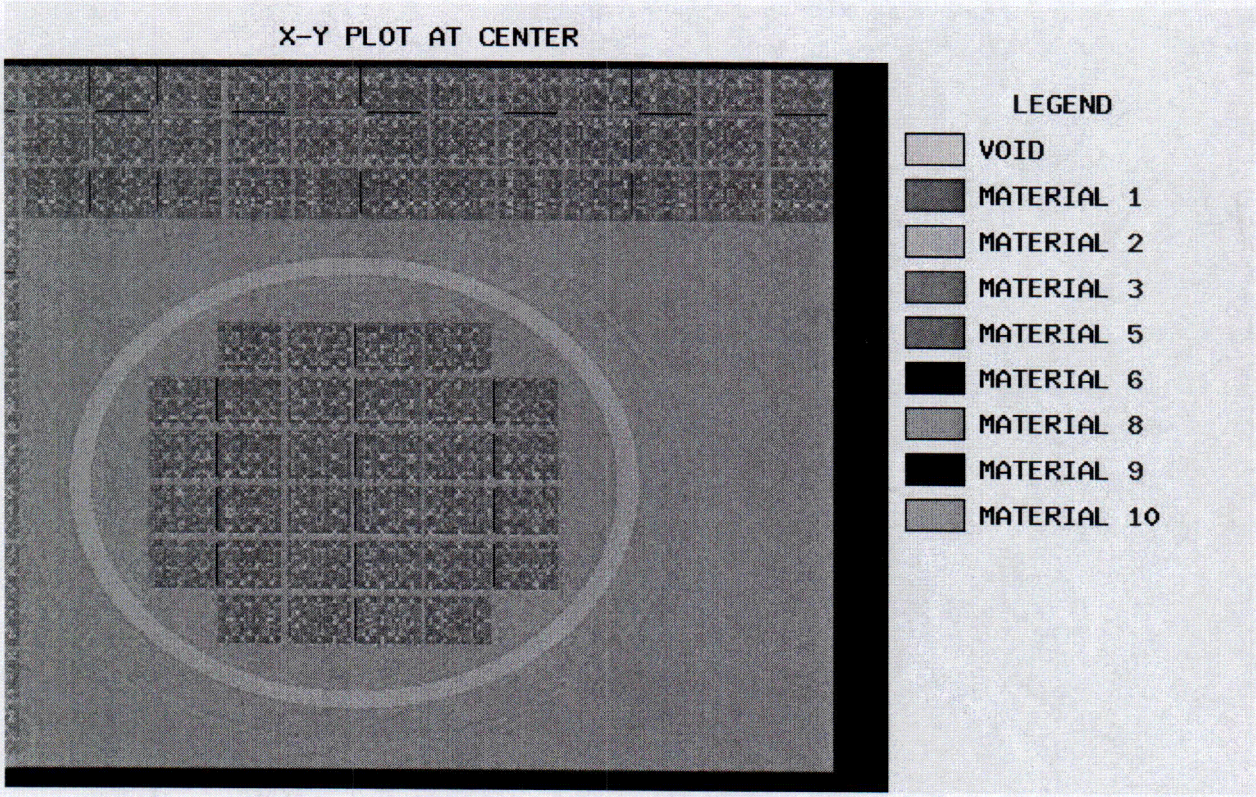
The reactivity effect of an abnormal heat load was also analyzed. Temperatures from near freezing to boiling (4°C through 100°C) were modeled to assess the effect of pool temperature on reactivity. Increased pool temperature reduces the moderator density and the density of soluble boron. For the finite geometry modeled as the reference case, increasing the moderator and fuel temperatures, as well as reducing the moderator densities with and without soluble boron, had a slightly negative (less than 0.006 reduction in k_{eff}) effect on reactivity.

6.5 Seismic Event

The reactivity effect of a postulated seismic event was also analyzed. This scenario assumes that all bundles within the MPC are seismically shifted to an off-center position at the maximum distance relative to their nominal centered position within the MPC storage cells. As a result, four assemblies within each two by two array of adjacent storage cells are assumed to be flush in the corners along the central interior intersection formed by the cell walls common to all four cells as shown in Figure 6-4. The configuration with all assemblies centered within the storage cells results in the maximum neutronic coupling between neighboring assemblies, and therefore the maximum position related reactivity. The configuration wherein assemblies are seismically shifted as described results in a slightly reduced reactivity (less than 0.0003 reduction in k_{eff}). 1

6.6 Fresh Fuel Assembly Mis-loaded into the MPC

The reactivity effect of a fresh fuel assembly misloaded into the MPC was analyzed. The two scenarios analyzed were: a fresh fuel assembly mis-loaded on the periphery of the MPC, as well as a fresh assembly mis-loaded into a central storage cell. For the limiting case, a fresh fuel assembly misloaded into a central storage cell, an additional 121 ppm of soluble boron is required to maintain $k_{eff} \leq 0.95$. The soluble boron concentration required to accommodate the presence of the MPC, now becomes 371 ppm (250 ppm to maintain $k_{eff} \leq 0.95$ under normal conditions and an additional 121 ppm to accommodate the limiting accident). A soluble boron concentration of 371 ppm is well below the current Technical Specification limit of 786 ppm. 2

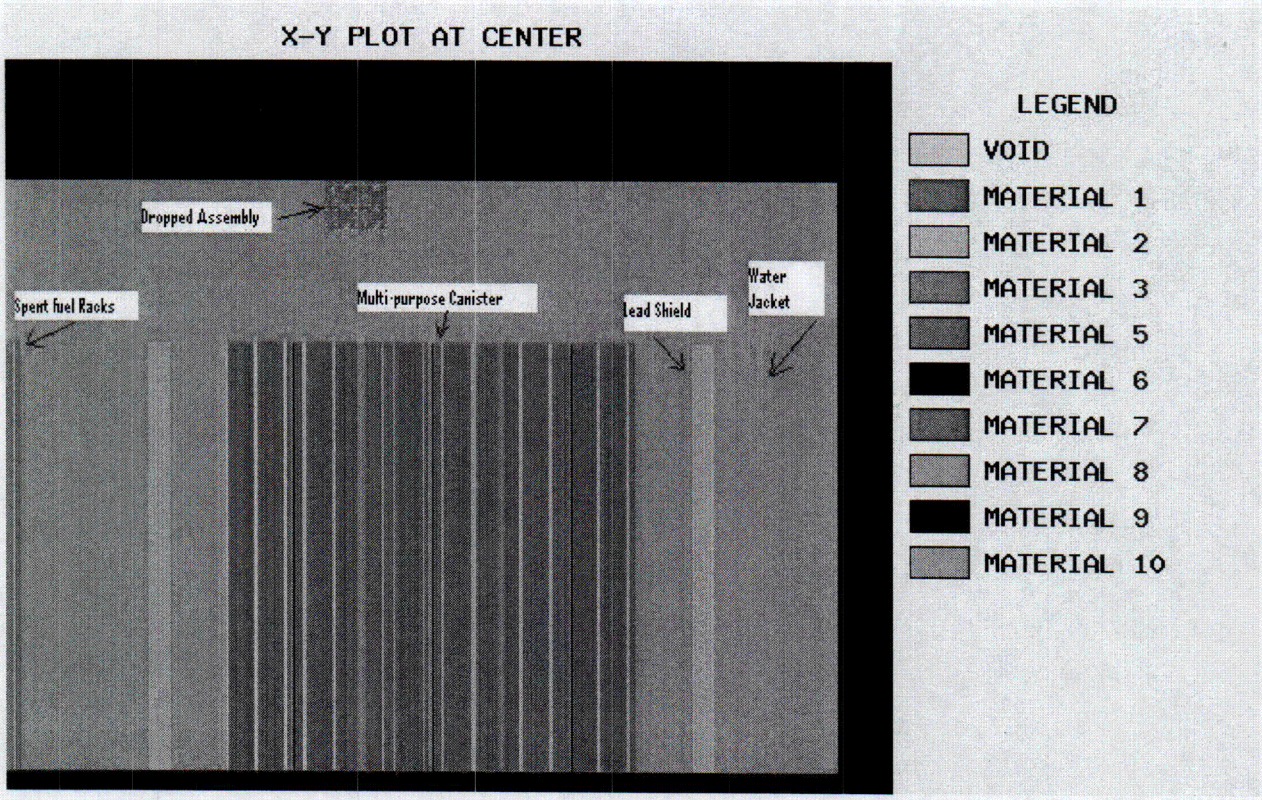


Key:

Material Number	Material
1	UO_2 (1.8 w/o)
2	Zircaloy
3	Unborated Water
5	Stainless Steel
6	Boraflex
8	Lead
9	Boron Carbide
10	Aluminum

Figure 6-1: Two Dimensional Cross Section Plot of a Fully Loaded MPC-32 in the Northeast Corner of the Indian Point 2 Cask Pit

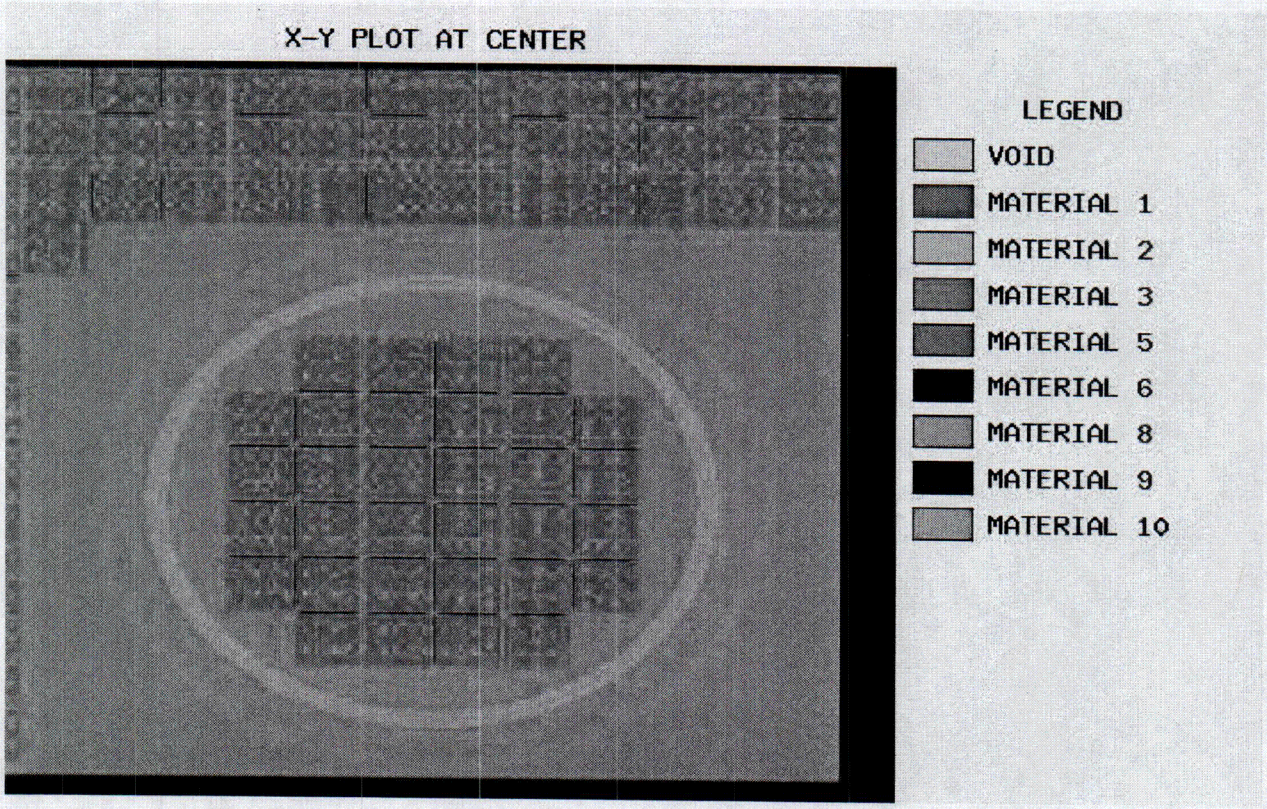
(NOTE: Some detail of model geometry is not apparent due to plot resolution limit.)



Key:

Material Number	Material
1	UO_2 (1.8 w/o)
2	Zircaloy
3	Unborated Water
5	Stainless Steel
6	Boraflex
7	UO_2 (Dropped Assembly)
8	Lead
9	Boron Carbide
10	Aluminum

Figure 6-2: Two Dimensional Axial Plot of Loaded MPC-32 Centered in the Indian Point 2 Cask Pit with Fully a Loaded Assembly Resting on Top



Key:

Material Number	Material
1	UO_2 (1.8 w/o)
2	Zircaloy
3	Unborated Water
5	Stainless Steel
6	Boraflex
8	Lead
9	Boron Carbide
10	Aluminum

Figure 6-3: Two Dimensional Cross Sectional Plot of Loaded MPC-32 in the Indian Point 2 Cask Pit with a Dropped Assembly Resting Adjacent to Spent Fuel Storage Racks

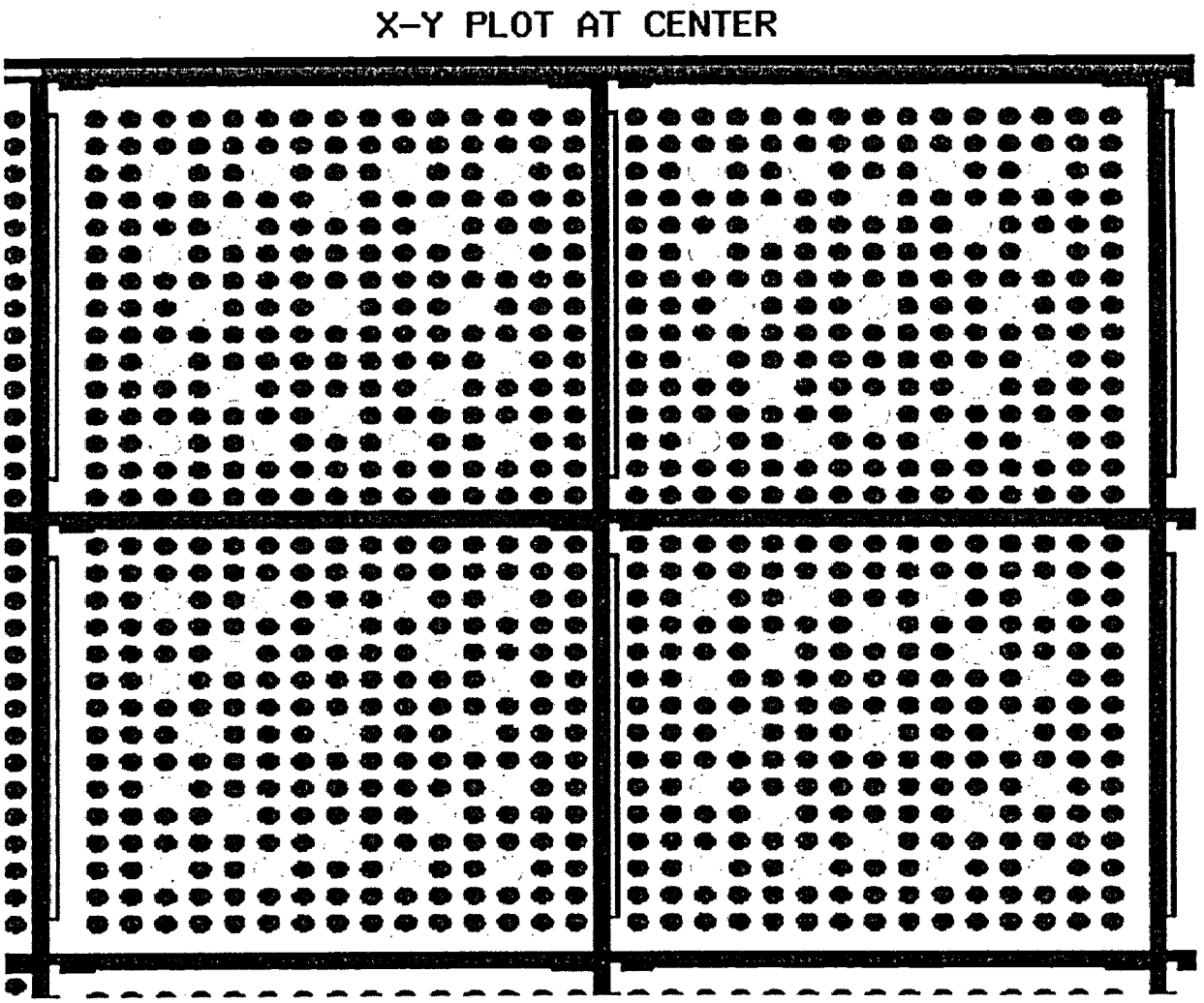


Figure 6-4: Adjacent Assemblies Shifted Toward Central Interior Intersection of Two-by-Two Storage Cell Array Walls

7.0 Conclusions

The analyses contained herein demonstrate that the proposed License Amendment Request satisfies the requirements of 10CFR 50.68, as applicable to cask loading operations in the IPEC Unit 2 spent fuel storage pool, in that such operations do not present an undue risk to the health and safety of the public. The conservative analyses presented illustrate that the fully loaded MPC will remain subcritical under the most reactive conditions, including accident and off-normal conditions. In addition, soluble boron requirements previously determined are sufficient to maintain $k_{eff} \leq 0.95$ during cask loading activities, therefore current pool Technical Specification limits for soluble boron remain valid. 2

8.0 References

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ATTACHMENT 4 TO NL-06-020

BENCHMARKING COMPUTER CODES FOR CALCULATING THE REACTIVITY STATE OF SPENT FUEL STORAGE RACKS, STORAGE CASKS AND TRANSPORTATION CASKS

Entergy Nuclear Operations, Inc.
Indian Point Unit No. 2
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Report No. 901-02-05

Benchmarking Computer Codes for Calculating the Reactivity State of Spent Fuel Storage Racks, Storage Casks and Transportation Casks

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Review/Approval Record

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1.0 Introduction

This report documents the results of benchmark calculations of three computer codes used to compute the reactivity state of nuclear fuel assemblies in close-packed arrays. Such close-packed arrays include high density spent fuel storage racks, dry storage casks and casks for transporting nuclear fuel. The three computer codes, which were benchmarked and validated are:

- KENO V.a, which is a module of SCALE 5[1]
- MCNP5[2]
- CASMO-4[3]

Earlier versions of KENO and CASMO have been previously benchmarked and validated by NETCO.[4,5]

To benchmark and validate the codes for spent fuel racks and cask evaluations, KENO and MCNP were used to simulate a series of critical experiments. The calculated eigenvalues (k_{eff}) were then compared with the critical condition (k_{eff} = 1.0) to determine the bias inherent in the calculated values. For the KENO V.a calculation, the 238 energy group ENDF/B-V cross-section library was used. For the MCNP5 calculations, the continuous energy cross-section library, based on ENDF/B-VI, was used.

After determining the inherent biases associated with KENO V.a and MCNP5, both KENO V.a and CASMO-4 (with its own 70 energy group cross-section library) were used to model central arrays of select critical experiments. It is noted that CASMO-4 models an infinitely repeating array of fuel assemblies and is generally used to generate cross-sections for core simulator models. As such, it does not lend itself directly to finite arrays of fuel racks surrounded by a reflector, as is the case in the critical experiments considered. Accordingly, the central fuel arrays of five critical experiments were modeled as infinite arrays with both KENO V.a and CASMO-4. A comparison of the KENO V.a and CASMO-4 eigenvalues provides a means to determine the CASMO-4 bias.

For the purposes of benchmarking, a set of five Babcock and Wilcox (B&W) critical experiments (XIII, XIV, XV, XVII, and XIX)[6] were selected, because they closely represent typical fuel/rack geometries with neutron absorber panels. In addition, the International Committee for Safety of Nuclear Installations (CSNI) identified a sequence of benchmark problems[7] that closely replicate both fuel/rack and fuel/cask geometries, and include typical light water reactor (LWR) enrichments and $H/^{235}U$ ratios. The resulting models are representative of most fuel storage rack and fuel cask configurations used today.

All work completed for the benchmarking calculations was carried out under NETCO's Quality Assurance Program[8]. The methods employed have been patterned to comply with accepted industry standards[9,10,11] and with accepted industry criticality references[12, 13, 14, 15, 16].

2.0 BENCHMARKING - STANDARD PROBLEMS AND CONFIGURATION CONTROL

2.1 SCALE-5 and MCNP5 Configuration Control

The binary executable codes and associated batch files were provided by RSICC on CD-ROM for use on Intel Pentium based micro-computers running under the Windows operating system. In this form, the programs can not be altered or modified. In addition to the binary executable codes, there are several supporting files which contain cross-section sets, etc. The file name, file size, and creation date for each executable file is given in Appendix A. Prior to executing either code sequence, the user will verify the file names, creation dates, and sizes to ensure that they have not been changed. Appendix B contains two CD-ROMs, which include the as-received versions of all files required to execute these programs. In all applications described in this report and for all subsequent applications, the files listed in Appendix A are to be used. (This appendix is not provided in the non-proprietary version of this report.)*

2.2 Sample Problems

A suite of input files with their corresponding output files were provided with each code. The input file names and batch files used to execute them are listed in Appendix A. These were executed on NETCO's host computer via batch files provided by RSICC and the resulting output files compared to those provided by RSICC on CD-ROM. Except for the date and time of execution stamps, the respective output files were identical. Each code uses a pseudo-random number generator that is initiated with a default seed value. Since the default value was used in each case, the sequences of random numbers were the same, leading to identical calculations. This verifies that the as-received versions of both codes are identical to the versions documented in the User's Manuals[1,2].

*Appendices A, B, C, D and E are included in the proprietary version of this report.

Examination of the sample input decks shows that the run modules in batch files exercise all of the code options used by this benchmarking exercise. Before and after each subsequent use of each code, one set of sample input modules are executed and the output files compared to the sample output files to verify that no system degradation has occurred. (All of these files are contained in Appendix B at the end of this report.)* This appendix is not provided in the non-proprietary version of this report.

2.3 CASMO-4 Configuration Control

The version of CASMO-4 used for these analyses was developed for a RISC workstation. Version 2.05.01 of CASMO-4 was used for this benchmarking work. For subsequent usage of CASMO-4 by NETCO, it shall be verified that Version 2.05.01 is being used. All versions of CASMO-4 are controlled by Studsvik of America under their Quality Assurance Program[17]. If a different version of CASMO-4 were to be used by NETCO for any subsequent analyses, the CASMO-4 analyses in Section 3.2 would be repeated with the version then in use.

*Appendices A, B, C, D and E are included in the proprietary version of this report.

3.0 BENCHMARK MODELING OF LWR CRITICAL EXPERIMENTS

An index of input and output files for each experiment modeled is contained in Appendices C and D. For each experiment, the input and output files are on 3.5 inch 1.44 MB diskettes which are also contained in Appendices C and D. Appendix E contains the calculation notebook for this project and represents a permanent record of all hand calculations performed during input preparation. All input parameters are fully traceable to the appropriate source documents. (These appendices are not provided in the non-proprietary version of this report.)*

3.1 BENCHMARKING OF SCALE-5 and MCNP5

The B&W experiments[6] include twenty (20) water moderated LWR fuel rod cores and close-packed critical LWR fuel storage arrays. Of these, five (5) used boron carbide/aluminum cermet poison plates (BORAL) in the closest possible packing geometry representing a 3 x 3 array of LWR fuel assemblies in high density fuel storage racks. These five (5) experiments have been modeled, as they most closely represent LWR fuel in high density fuel storage racks and cask configurations with neutron absorber panels. Table 3-1 summarizes some of the parameters used in the models, including U-235 enrichment, moderator-to-fuel ratio and absorber macroscopic absorption cross-section.

The Committee for Safety of Nuclear Installations (CSNI) has published a selection of critical experiments[7], which are a sequence of exercises arranged in order of increasing complexity, introducing one new parameter into the geometry and materials at a time. They were selected specifically to validate calculational methods for criticality safety assessments. The fuel is designed to simulate LWR fuel, is water moderated, and the lattices include BORAL plates between assemblies when neutron poisons are

*Appendices A, B, C, D and E are included in the proprietary version of this report.

included. The sequence starts with Experiment 1-1, a single array of 20 x 18, 2.35 w/o ^{235}U rods with a water reflector all around. Experiments 1-2-1 and 1-2-2 are also single water reflected arrays but are at a higher enrichment (4.74 w/o ^{235}U) and are at undermoderated (1-2-1) and optimum moderation (1-2-2) conditions. Experiment 2-1 has three square arrays of 2.35 w/o ^{235}U fuel separated by BORAL neutron absorber plates. Experiment 2-2 has a 2 x 2 array of four 4.74 w/o ^{235}U rod arrays also separated by BORAL plates. Experiments 3-A-1 and 3-B-1 are similar to experiment 2-1 but include, respectively, lead and steel reflecting walls. Experiment 3-A-2 is similar to Experiment 2-2 but also has a lead reflecting wall.

In each MCNP5 model of the criticals, 4,000,000 neutrons in 2,000 generations were tracked. In each KENO model of the criticals, at least 20,000,000 neutrons in at least 10,000 generations were tracked. The output files were always checked to ensure that the fission source distribution had converged. A summary of the distribution of k_{eff} over all generations is automatically plotted in the output files and shows them to be approximately normally distributed. Thus, normal one-sided tolerance limits with appropriate 95% probability / 95% confidence factors (95/95) can be used. The calculated results for each critical experiment are presented in Table 3-2, including the calculated k_{eff}, the one-standard-deviation statistical uncertainty of k_{eff}, denoted by σ, and the bias with respect to the critical state k_{eff} = 1.0.

The overall bias between the calculation eigenvalue and the experiments is calculated as follows. First, the variance-weighted mean is calculated as

$$k_m = \sum_{i=1}^{N} (k_i / \sigma_i^2) / \sum_{i=1}^{N} (1/\sigma_i^2) \qquad (3\text{-}1)$$

where N = 13 (for the 5 B&W and 8 CSNI criticals), k_i is the SCALE-5 calculated k_{eff} for critical i, and σ_i is the SCALE-5 calculated standard deviation of the distribution of k_{eff} for critical i. The standard deviation around k_m is given by

$$\sigma_m = \left[\frac{1}{N-1} \sum_{i=1}^{N} (k_i - k_m)^2 \right]^{1/2} \qquad (3\text{-}2)$$

The bias is calculated as k_m - 1, and has the same standard deviation as k_m. Based upon the results shown in Table 3-2, it is recommended that the 238 energy group ENDF/B-V library be used in all criticality analyses. For SCALE-5, the resulting mean bias for this library is -0.00782 ± 0.00361. For MCNP5, using the continuous energy cross-section library based on ENDF/B-VI, the resulting variance weighted mean bias is - 0.00574 ± 0.00509.

Correlations of bias with respect to moderator-to-fuel ratio (H / ^{235}U), number density ratio and absorber strength (Σ_a^{th}) were investigated and found to be not significant. The coefficient of determination for bias versus moderator-to-fuel ratio for the 238 group ENDF/B-V library was a negligible 2.6%, whereas for MCNP5 it was 4.1%, indicating that the method bias is not strongly dependent on moderator-to-fuel ratio. In all cases, the bias becomes less negative with decreasing moderator-to-fuel ratio (i.e., increasing enrichment). The coefficient of determination for bias versus absorber strength for the 238 Group ENDF/B-V library was an insignificant 6.1%, while for MCNP5, it was 37.1%. In all cases, the bias becomes less negative with increased absorber strength. These results are illustrated in Figures 3-1 and 3-2, respectively.

Table 3-1: B&W [6] and CSNI [7] Critical Experiments - Design Parameters

Reference	Experiment Number	Absorber Type	Absorber Σ_a [cm^{-1}]	Enrichment w/o	$H/^{235}U$ Ratio
6	XIII	BORAL	1.871	2.459	216.43
6	XIV	BORAL	1.460	2.459	216.52
6	XV	BORAL	0.475	2.459	216.52
6	XVII	BORAL	0.293	2.459	216.54
6	XIX	BORAL	0.129	2.459	216.54
7	1-1	none	-	2.35	398.72
7	1-2-1	none	-	4.75	109.44
7	1-2-2	none	-	4.75	228.53
7	2-1	BORAL	30.6	2.35	398.72
7	2-2	BORAL	24.6	4.75	228.53
7	3-A-1	none	-	2.35	398.75
7	3-B-1	none	-	2.35	398.75
7	3-A-2	BORAL	24.6	4.75	228.53

Table 3-2 B&W[6] and CSNI[7] Critical Experiment Results

Reference	Experiment		KENO V.a			MCNP5	
		k_{eff}	σ	bias	k_{eff}	σ	bias
6	XIII	0.99341	0.00017	-0.00659	0.99422	0.00035	-0.00578
6	XIV	0.98989	0.00018	-0.01011	0.98997	0.00035	-0.01003
6	XV	0.98623	0.00017	-0.01377	0.98525	0.00035	-0.01475
6	XVII	0.98972	0.00016	-0.01028	0.98846	0.00034	-0.01154
6	XIX	0.99136	0.00018	-0.00864	0.99004	0.00035	-0.00996
7	1-1	0.99048	0.00017	-0.00952	0.99294	0.00032	-0.00706
7	1-2-1	0.99404	0.00020	-0.00596	1.00000	0.00030	0.00000
7	1-2-2	0.99774	0.00020	-0.00226	1.00000	0.00030	0.00000
7	2-1	0.98925	0.00017	-0.01075	0.99164	0.00032	-0.00836
7	2-2	0.99549	0.00020	-0.00451	1.00000	0.00030	0.00000
7	3-A-1	0.99390	0.00018	-0.00610	0.99012	0.00033	-0.00988
7	3-B-1	0.99287	0.00017	-0.00713	0.99590	0.00033	-0.00410
7	3-A-2	0.99904	0.00020	-0.00096	1.00000	0.00030	0.00000
	Arithmetic Mean	0.99218			0.99426		
	Variance Weighted			-0.00782			-0.00574
	Standard Deviation			±0.00361			±0.00509

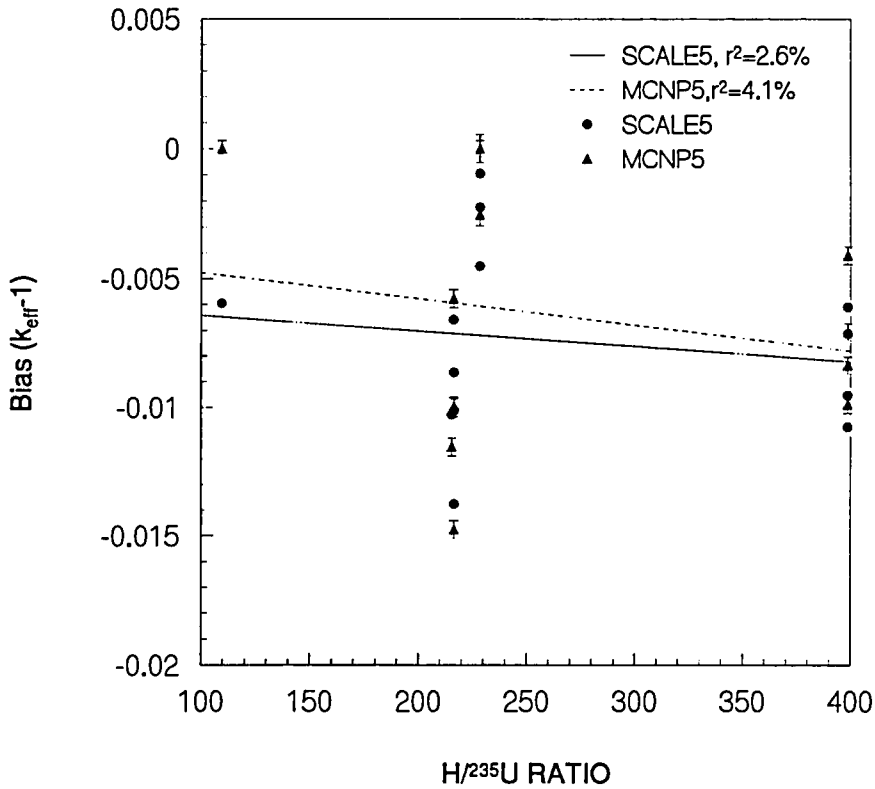

Figure 3-1: Variation of Bias (k_{eff}-1) with Moderator-to-Fuel Ratio

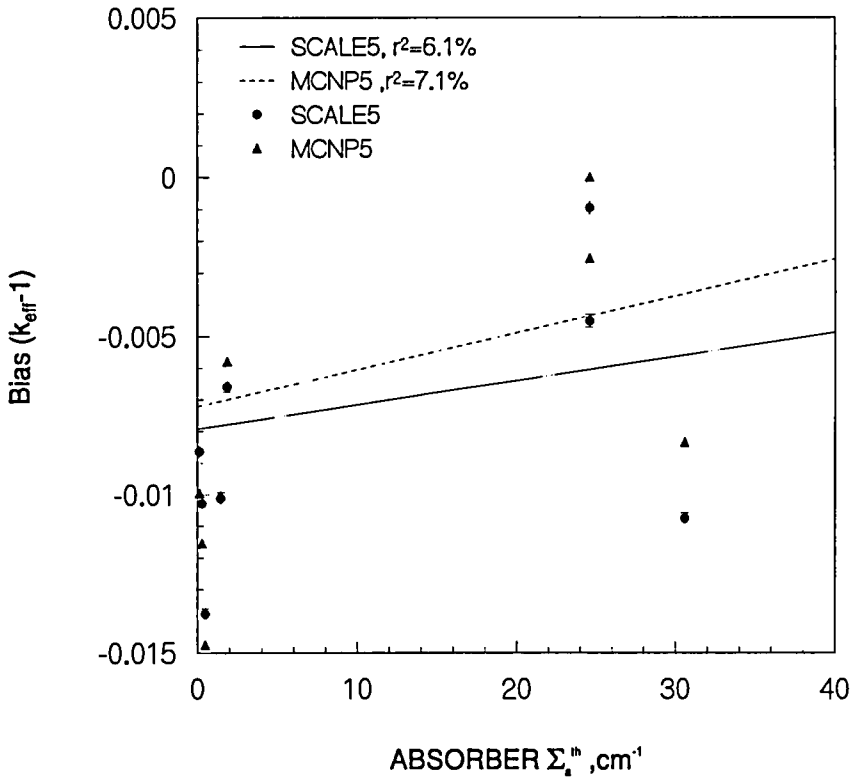

Figure 3-2: Variation of Bias (k_{eff}-1) with Absorber Strength

3.2 BENCHMARKING OF CASMO-4

This section compares SCALE-5[1] and CASMO-4[3] calculations for k_∞ of the same five B&W critical experiments[6] discussed in Section 3.1. CASMO-4 is limited in its ability to render a geometric model and can only be used for infinite arrays of assemblies. Thus, for this benchmark analysis, the central assembly of the 3 x 3 array of assemblies in the B&W critical experiments was modeled and then assumed to be infinitely reflected. The assembly pitch was preserved in the model, but the effect of the finite water reflector around the 3 x 3 array was lost, making the model supercritical.

SCALE-5 was also used to model the B&W critical experiments with exactly the same geometry as they were rendered in CASMO-4. Because the bias of SCALE-5 is known (see Section 3.1), it can be applied to the SCALE-5 result to obtain a best-estimate of the supercritical state of the infinitely reflected assembly model. The CASMO-4 result can then be compared with this best estimate to obtain a CASMO-4 bias.

The results of the SCALE-5 and CASMO-4 analyses are compared in Table 3-3. The CASMO-4 bias is calculated as

$$\text{bias}_{\text{CASMO-4}} = k_{\text{CASMO-4}} - k_{\text{SCALE-5, best estimate}}$$

where

$$k_{\text{SCALE-5, best estimate}} = k_{\text{SCALE-5}} - \text{bias}_{\text{SCALE-5}}$$

For CASMO-4 the resulting mean bias and standard deviation for the 238 Group ENDF/B-V library are -0.01028 and 0.00198, respectively.

Table 3-3: B&W Critical Experiments as CASMO Infinite Arrays - Results

Experiment	CASMO-4 k_∞	SCALE PC(bias corrected) 238 GROUP NDF 5		
		k_∞	σ	bias
XIII	1.08947	1.10160	0.00050	-0.01423
XIV	1.08993	1.10175	0.00049	-0.01523
XV	1.09898	1.10961	0.00045	-0.01280
XVII	1.10770	1.11732	0.00045	-0.00945
XIX	1.11607	1.12330	0.00043	-0.00832
			bias	-0.01028
			σ	0.00198

4.0 CONCLUSIONS

SCALE-5 and MCNP5 have been benchmarked by modeling five (5) Babcock and Wilcox critical experiments and eight (8) CSNI critical experiments representative of fuel storage rack and fuel cask geometries. At a 95% probability / 95% confidence level, the computed bias for SCALE-5 and MCNP5 are -0.01381 and -0.01460, respectively.

CASMO-4 has also been benchmarked by modeling the five (5) Babcock and Wilcox critical experiments as infinite arrays. Best estimates of k_∞ for the exact same geometry were calculated using SCALE-5 and applying the mean bias reported above. The CASMO-4 bias with respect to these values was calculated to be -0.01028 ± 0.00198 (1- sigma). The comparison of SCALE-5 and CASMO-4 serves to verify the results of each with respect to the other.

It is therefore concluded that these calculational methods have been adequately benchmarked and validated. They may be used individually or in combination for the criticality analysis of spent fuel storage racks, fuel casks and fuel casks in close proximity to fuel storage racks, provided the appropriate biases are applied.

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