Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Entergy Nuclear Indian Point 2, LLC
P. O. Box 249
Buchanan, NY 10511

September 20, 2001

Re: Indian Point Unit No. 2
Docket No. 50-247
NL 01-110

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop 0-P1-17
Washington, DC 20555-0001

SUBJECT: License Amendment Request (LAR 01-010) for Spent Fuel Storage Pit Rack Criticality Analysis with Soluble Boron Credit

References:
1. NRC letter to Con Edison (RA 90-076), titled "Issuance of Amendment 150 (TAC No. 72962)," dated April 19, 1990
2. Licensee Event Report 2000-004-00, titled "Design Basis Compliance Failure Due to Spent Fuel Storage Rack Boraflex Degradation," dated May 4, 2000

Pursuant to 10CFR50.90, Entergy Nuclear Operations, Inc. (ENO) hereby requests the following amendment to the Indian Point Unit No. 2 (IP2) Technical Specifications (TS) Section 3.8, "Refueling, Fuel Storage and Operations with the Reactor Vessel Head Bolts Less Than Fully Tensioned," Table 4.1-2, "Frequencies for Sampling Tests," and Section 5.4, "Fuel Storage." The purpose of this License Amendment Request is to allow credit for soluble boron in the Spent Fuel Pit (SFP) criticality analysis. IP2 License Amendment 150 authorized an increase to the storage capacity of the SFP using high-density storage racks containing Boraflex as a neutron absorber. The criticality analysis supporting License Amendment 150 assumed that the SFP water would be unborated. In Licensee Event Report 2000-004-00, IP2 reported Boraflex degradation and implemented a "checkerboard" fuel distribution pattern to conservatively maintain criticality margins. This action effectively reduced the capacity of the SFP. IP2 committed to pursue SFP criticality credit for soluble boron using NRC approved methodology as a long-term solution to restore the capacity of the SFP.

Attachment 1 to this letter provides the description and evaluation of the proposed change. The revised TS pages and TS Bases pages are provided in Attachment 2 (strikeout and shadow format). Attachment 3 provides the Northeast Technology Corporation Report NET-173-01, "Criticality Analysis for Soluble Boron and Burnup Credit in the Con Edison Indian Point Unit No. 2 Spent Fuel Storage Racks." Attachment 4 provides the Northeast Technology

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Corporation Report NET-173-02, "Indian Point Unit 2 Spent Fuel Pool (SFP) Boron Dilution Analysis."

Restoration of the SFP capacity is essential to the conduct of the refueling outage scheduled for the Fall of 2002. ENO requests approval of the proposed change by May 31, 2002 with an implementation date within 60 days of approval.

The Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC) have reviewed the proposed change. Both committees concur that the proposed change does not involve a significant hazards consideration as defined by 10 CFR 50.92(c).

In accordance with 10 CFR 50.91, a copy of this submittal and the associated attachments are being submitted to the designated New York State official.

This letter contains no new commitments.

Should you or your staff have any questions regarding this submittal, please contact Mr. John F. McCann, Manager, Nuclear Safety and Licensing at (914) 734-5074.

Very truly yours,

Fred Dacimo
Vice President – Operations
Indian Point 2

cc:

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-247
Indian Point Nuclear Generating Unit No. 2	)	

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission (NRC), Entergy Nuclear Operations, Inc., as holder of Facility Operating License No. DPR-26, hereby applies for amendment of the Technical Specifications contained in Appendix A of this license.

The specific proposed Technical Specification revision is set forth in Attachment 2. The associated assessment demonstrates that the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c).

As required by 10CFR50.91(b)(1), a copy of this Application and our evaluation concluding that the proposed change does not involve a significant hazards consideration has been provided to the appropriate New York State official designated to receive such amendments.

BY: _____
Fred Dacimo
Vice President – Operations
Indian Point 2

Subscribed and sworn to
before me this 20 day
September, 2001.

Ersilia A. Amanna (Bowie)
Notary Public

ERSILIA A. AMANNA
Notary Public, State of New York
No. 01AM6038689
Qualified in Westchester County
Commission Expires March 20, 2002

ATTACHMENT 1 TO NL 01-110

LICENSE AMENDMENT REQUEST

SPENT FUEL STORAGE PIT RACK CRITICALITY ANALYSIS WITH SOLUBLE BORON CREDIT

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

LICENSE AMENDMENT REQUEST

DESCRIPTION OF THE PROPOSED CHANGE

Entergy Nuclear Operations, Inc. (ENO) is requesting a change to the Indian Point Unit No. 2 (IP2) Technical Specifications (TS) as described below.

The proposed change affects:

- TS 3.8.D that imposes Limiting Conditions for Operation for the Spent Fuel Pit (SFP) anytime it contains fuel.
- Table 4.1-2, that imposes frequencies for sampling tests.
- TS 5.4, Fuel Storage Design Features, that presently states that the spent fuel storage racks are designed and their loading maintained such that $k_{eff} \leq 0.95$ even if unborated water were used to fill the pit and with fuel assemblies containing a maximum enrichment of 5.0 weight percent ($^{w}/o$) ^{235}U.

REASONS FOR THE CHANGE

On May 4, 2000 (Ref. 1), IP2 reported to the NRC that the assumptions used in the criticality analysis for the SFP were no longer met as a result of the discovery of Boraflex thinning and gaps. IP2 removed spent fuel assemblies from selected storage rack cells to implement a "checkerboard" fuel distribution pattern that is bounded by a criticality analysis assuming unborated pit water. However, this short-term corrective action reduced the capacity of the SFP. This reduced capacity will impact the ability to efficiently implement refueling outages (RFO), including the RFO scheduled for the Fall of 2002. Therefore, IP2 committed to long-term corrective actions to restore the SFP capacity. One of those long-term corrective actions, a licensing action to obtain credit for soluble boron in the SFP, is the subject of this Licensing Action Request (LAR).

Approval of this proposed licensing action will restore the capacity of the SFP to support continued efficient IP2 operations.

EVALUATION OF THE PROPOSED CHANGE

At IP2 spent fuel and new fuel ready for insertion into the reactor are stored in the SFP. Storage racks are provided to hold spent fuel assemblies and are erected on the SFP floor. The fuel storage racks are high-density racks that have been designed to provide a maximum storage capacity of 1376 locations. There are two types of storage rack arrays:

- Region I, consisting of three racks, has 269 storage locations for fresh or irradiated fuel assemblies with enrichments up to 5.0 $^{w}/o$ ^{235}U. Each Region I storage cell is a square box with an 8.75 inch inside dimension. Boraflex poison is held in place adjacent to each side of the box

by "picture-frame" sheathing. The boxes are assembled into racks with an east-west pitch of 10.765 inches (center-to-center) and a north-south pitch of 10.545 inches.

- Region II, consisting of nine racks, can store 1105 fuel assemblies with enrichments and burnups as specified in the TS. Region II racks consist of boxes welded into a "checkerboard" array with a storage location in each square. One Boraflex poison strip is held to one side of each cell wall by picture frame sheathing. Two failed fuel canisters can also be stored.

The licensing bases criterion for IP2 for the prevention of fuel storage criticality is specified in UFSAR 9.5.1.1 as "Criticality in the new and spent fuel storage pits shall be prevented by physical means or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls." This criterion is currently implemented by storing fuel vertically in an array with sufficient center-to-center distance between assemblies to assure $k_{eff} \leq 0.95$ even if unborated water is used to fill the pit. The limits on enrichment and burnup of the fuel are specified in TS 3.8. This licensing basis is consistent with the current 10CFR50 Appendix A, Criterion 62, "Prevention of Criticality in Fuel Storage and Handling," and NUREG-0800 paragraph 9.1.2, "Spent Fuel Storage."

In 1990 (Ref. 2), the IP2 Facility Operating License was amended to permit reracking of the SFP with high-density storage racks containing Boraflex as a neutron absorber. The high density storage racks increased the SFP storage capacity from 980 to 1376 fuel assemblies. The amendment also increased the allowable fuel enrichment from 4.3 $^{w}/o$ ^{235}U to 5.0 $^{w}/o$ ^{235}U. The criticality analysis supporting the reracking conservatively assumed that the SFP water was unborated and that there would be no formation of Boraflex gaps. License Amendment 150 approval was based upon a licensee long-term surveillance program to monitor the performance of the Boraflex in the SFP.

In 1996 (Ref. 3) in response to GL 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," IP2 reported that the physical condition of the Boraflex in the IP2 storage racks was acceptable. In 2000 (Ref. 1) as a result of a monitoring program inspection, IP2 reported to the NRC that some Boraflex panels had developed gaps. As a short-term corrective action, IP2 implemented a "checkerboard" fuel distribution pattern that was bounded by a criticality analysis assuming unborated pit water.

This LAR, if approved, would implement a LER long-term corrective action that is consistent with 10CFR50.68, "Criticality Accident Requirements." In accordance with 10CFR50.68(a), ENO has elected to comply with 10CFR50.68(b). 10CFR50.68(b)(4) states:

> " ... If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

The proposed TS, as demonstrated by the analyses (Attachments 3 and 4), ensure that the criteria of 10CFR50.68(b) are met for the IP2 SFP.

In order to mitigate the effects of degraded Boraflex, each of the two SFP regions will be further subdivided into two sub-regions with Limiting Conditions for Operation proposed for each sub-region. One sub-region in each region takes credit for remaining Boraflex; the other sub-region takes no credit for Boraflex.

- Region 1-1 takes no credit for Boraflex and can safely accommodate assemblies that have been discharged with minimum burnup. No credit for ^{241}Pu decay is taken in Region 1-1. Region 1-1 can accommodate unirradiated fuel up to 5.0 ^{w}/o ^{235}U, assuming a minimum number of Integral Fuel Burnable Absorbers (IFBA) using a 1 of 2 checkerboard loading pattern with the alternate cells left vacant.
- Region 1-2 is assumed to have sustained a 50% loss of Boraflex. Region 1-2 can accommodate unirradiated fuel up to 5.0 ^{w}/o ^{235}U assuming a minimum number of IFBA rods.
- Region 2-1 takes no credit for Boraflex and can accommodate assemblies that have been discharged with a minimum burnup and have cooled for a minimum amount of time.
- Region 2-2 is assumed to have sustained a 30% loss of Boraflex. Region 2-2 can accommodate assemblies that have been discharged with a minimum burnup and have cooled for a minimum amount of time.

All fuel moves are planned and reviewed in advance. The addition of two more sub-regions does not pose a significant burden or challenge to refueling personnel.

Criticality Analysis Methodology

Attachment 3, Northeast Technology Corporation report NET-173-01, "Criticality Analysis for Soluble Boron and Burnup Credit in the Con Edison Indian Point Unit No. 2 Spent Fuel Storage Racks," provides the detailed description of the methodology used to determine the required soluble boron concentrations for accident and non-accident conditions and to predict the degradation of Boraflex.

The analysis for Boraflex credit is based on testing performed in 2000. The analysis shows that the credit taken for Boraflex is conservative through the end of 2006. In areas where 50% Boraflex credit is taken, the predicted bounding loss is 44.2%. In areas where 70% credit is taken, the predicted bounding loss is 21.6%.

The analysis shows that a minimum soluble boron concentration of 786 ppm will assure the ≤ 0.95 k_{eff} requirement under non-accident conditions. The analysis shows that if, in the unlikely event all soluble boron was lost, k_{eff} would remain <1.0.

The analysis considered the effects of 1.) a dropped fuel assembly or an assembly placed alongside a rack, 2.) a misloaded fuel assembly, and 3.) abnormal heat loads. The analysis shows that, in the worst-case accident scenario, a boron concentration of 1495 ppm will assure the ≤ 0.95 k_{eff} requirement.

A conservative minimum TS requirement for SFP boron concentration of 2000 ppm is proposed. This requirement provides a large margin to the criticality analysis requirements. The limit is consistent with other TS requirements.[1] Selection of this limit thus simplifies plant operation.

Boron Dilution Analysis

ENO has performed an Indian Point 2 plant-specific SFP boron dilution analysis. Attachment 4, Northeast Technology Corporation report NET-173-02, "Indian Point Unit 2 Spent Fuel Pool (SFP) Boron Dilution Analysis," provides the detailed description of the methodology used to confirm the conservatism of the proposed minimum TS limit of 2000 ppm soluble boron concentration under postulated unplanned SFP dilution scenarios.

The analysis determined the dilution times and volumes necessary to dilute the SFP from an initial boron concentration of 2000 ppm to the minimum soluble boron concentration (786 ppm) required to maintain $k_{eff} \leq 0.95$. A volume of 230,551 gallons is required for this dilution. For the postulated worst case scenario, dilution from maximum flow of two primary water pumps, there are suitable methods of detection. And an acceptably large time, approximately 35.6 hours, would be available for plant personnel to detect and mitigate the dilution prior to reaching the low boron concentration limit.

Analyses' Conclusions

With the limits imposed on initial maximum enrichment and the establishment of minimum burnup criteria, ENO has concluded that:

- The value of k_{eff} will be less than 0.95 in the event of the worst-case accident when credit for a minimum boron concentration of 1495 ppm is taken.
- The value of k_{eff} will be less than 1.0 under normal operating conditions with a 95 percent probability, 95 percent confidence level when no credit for a minimum boron concentration is taken.
- An unplanned or inadvertent event that could dilute the SFP boron concentration from 2000 ppm to 786 ppm is not a credible event because of the low frequency of postulated initiating events and because the event would be readily detected and mitigated by plant personnel through alarms, flooding, and operator rounds through the SFP area.

Other Licenses

The NRC has recently issued an amendment (Ref. 4) to the Facility Operating Licenses for the McGuire Nuclear Station Units 1 and 2 that is similar to the License Amendment requested by ENO in this letter. In the McGuire amendment, credit was allowed for soluble boron and for a specified amount of Boraflex in specified regions.

[1] TS 3.3.A.1 requires the refueling water storage tank and the ESF accumulators to have a boron concentration of at least 2000 ppm. TS 3.8.B requires the reactor refueling water cavity to be filled with borated water with a concentration ≥ 2000 ppm under specified refueling conditions.

Overall Conclusion

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

Implementation

Plant processes and procedures will ensure that the limiting assumptions of these analyses are appropriately controlled so that they are not inadvertently exceeded. For example, operating procedures will ensure that, if the core offload time of 45 days will be exceeded, a re-analysis will be performed.

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

ENO has determined that this proposed Technical Specification change does not involve a significant hazard consideration as defined by 10CFR50.92(c).

1. **Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.**

   Current TS contain minimum requirements for the SFP boron concentration. The actual boron concentration in the SFP has been maintained at a higher value. The proposed changes to the TS establish new boron concentration requirements for the SFP water that are consistent with the new criticality analysis. Since soluble boron has already been maintained in the SFP water and is currently required by the TS, the implementation of this new requirement will have no effect on the normal SFP operations and maintenance.

   The presence of an increased requirement for soluble boron in the SFP water does not increase the probability of a fuel assembly drop accident in the SFP. The handling of the fuel assemblies in the SFP has always been performed in borated water. The criticality analysis shows the consequences of a fuel assembly drop accident in the SFP are not affected when considering the presence of soluble boron since the rack k_{eff} remains ≤ 0.95.

   Fuel assembly placement will continue to be controlled in accordance with approved fuel handling procedures and will be in accordance with TS spent fuel rack storage configuration limitations. The proposed SFP storage configuration limitations will be more complex but will be similar to those previously approved. Therefore, the new limitations will not significantly increase the probability of accident occurrence. There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel racks since the criticality analysis demonstrates that the SFP k_{eff} will remain ≤ 0.95 following an accidental misloading.

There is no increase in the probability of the loss of normal cooling to the spent fuel pit water when considering the presence of soluble boron in the pit water for subcriticality control since a high concentration of soluble boron has always been maintained in the SFP water.

Soluble boron requirements for mitigating reactivity effects due to increased pool temperatures are adequately met by the proposed increase in minimum TS soluble boron concentration. A negligible increase in the probability of a criticality accident due to increased pool temperature exists with the proposed TS changes, as the minimum soluble boron concentration will not change. The positive reactivity introduced as a result of the higher TS boron concentration effect on moderator reactivity coefficient will be sufficiently mitigated by the substantial margin to the amount actually required to maintain $k_{eff} \leq 0.95$.

Decreased fuel temperatures will increase the water density in the SFP, therefore increasing the thermal neutron flux, possibly causing an increase in reactivity. This density increase will increase the differential worth of the soluble boron but the excess soluble boron in the SFP is more than sufficient to offset any reactivity increase introduced by a temperature decrease.

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

2. **Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.**

Spent fuel handling accidents are not new or different types of accidents, they have been analyzed for the UFSAR and in Criticality Analysis reports associated with License Amendment 150 up to the nominal 5.0 $^{w}/o$ ^{235}U that is assumed for the proposed change.

A dilution of the SFP soluble boron has always been a possibility. However the boron dilution event previously had no consequences since boron was not previously credited. With the proposed TS, credit is taken for soluble boron. So a boron dilution has been evaluated as a possible new accident. The evaluation concluded a boron dilution accident was not credible, that processes were in place to detect and mitigate the possible events, and that, even if the SFP boron concentration was diluted to zero, criticality would not occur. Therefore, there would be no additional hazards if this request were approved.

There is no other change in the plant configuration or equipment design.

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

3. **Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.**

The TS changes proposed by this LAR and the resulting spent fuel storage operation limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. These limits are based on the plant specific criticality analysis and boron dilution analysis. The proposed TS changes rely upon known and predictable reactivity effects to ensure required criticality margins in the SFP.

While the criticality analysis utilizes credit for soluble boron, storage configurations have been defined using 95/95 k_{eff} calculations to ensure the spent fuel rack k_{eff} will be <1.0 with no soluble boron. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions and to provide subcritical margin such that the SFP k_{eff} is maintained ≤ 0.95.

The loss of substantial amounts of soluble boron from the SFP, which could lead to k_{eff} exceeding 0.95, has been evaluated and shown to be not credible. An evaluation has been performed that shows that the dilution of the SFP boron concentration from 2000 ppm to 786 ppm is not credible. Also the spent fuel rack k_{eff} will remain <1.0 with the SFP flooded with unborated water. These safety analyses demonstrate a level of safety comparable to the conservative criticality analysis approved for License Amendment 150 and show that the requirements of 10CFR50.68 are met.

The reactivity credit for additional poisons in the spent and fresh fuel assemblies increases the margin of safety in the SFP. No credit is taken for Boraflex in certain regions, when in reality some residual Boraflex does remain in these regions. In regions that do take credit for Boraflex, the amount of credit is conservative. These conservatisms add an increased safety margin. Predictions of the effective neutron multiplication factors have shown that, under the worst of scenarios, the SFP remains subcritical when conservative credit for future expected loss of Boraflex poison plates is considered.

The analysis show that the level of safety required by 10CFR50.68 is achieved for the IP2 SFP with the proposed TS.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

CONCLUSIONS

Based on the above evaluation, ENO has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed, and does not result in a reduction in any margin of safety. Therefore, operation of IP2 in accordance with the proposed amendment does not involve a significant hazards consideration. In addition, the

proposed change to the TS has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC). Both committees concur that the proposed change does not involve a significant hazards consideration.

ENVIRONMENTAL ASSESSMENT

An environmental assessment is not required for the above proposed change because the requested change to the Indian Point Unit No. 2 Technical Specifications conforms to the criteria for "actions eligible for categorical exclusion," as specified in 10CFR51.22(c)(9). The requested change will have no impact on the environment. The proposed change does not involve a significant hazards consideration as discussed in the preceding section. The proposed change does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed change does not involve a significant increase in individual or cumulative occupational radiation exposure.

References

1. Con Edison LER 2000-004-00 to NRC (NL 00-058), titled: "Design Basis Compliance Failure Due to Spent Fuel Storage Rack Boraflex Degradation," dated May 4, 2000
2. NRC letter to Con Edison (RA 90-076), titled: "Issuance of Amendment (TAC No. 72962)," dated April 19, 1990
3. Con Edison letter to NRC (NL96-0106), titled "Response to NRC Generic Letter 96-04: Boraflex Degradation in Spent Fuel Storage Racks," dated October 23, 1996
4. NRC letter to Duke Energy Corporation, titled "McGuire Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MA9730 and MA9731)," dated November 27, 2000

ATTACHMENT 2 TO NL 01-110

TECHNICAL SPECIFICATION PAGES IN

STRIKEOUT/SHADOW FORMAT

Deleted text is shown as ~~strikeout~~.

Added text is shown as shaded. New Figures are not shaded but are shown without an Amendment No.

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

LIST OF FIGURES

Title	Figure No.
Reactor Core Safety Limit-Four Loops In Operation	2.1-1
PORV Opening Pressure for Operation Less Than or Equal to 305°F	3.1.A-1
Maximum Pressurizer Level with PORVs Inoperable and One Charging Pump Energized	3.1.A-2
Maximum Reactor Coolant System Pressure for Operation With PORVs Inoperable and One Safety Injection Pump and/or Three Charging Pumps Energized	3.1.A-3
Coolant System Heatup Limitations	3.1.B-1
Coolant System Cooldown Limitations	3.1.B-2
~~DELETED~~ Spent Fuel Storage Rack Layout	3.8-1
~~Spent Fuel Storage Rack Layout~~ Region 1-1 Limiting Fuel Burnup versus Initial Enrichment	3.8-2
Region 1-2 ~~Limiting Fuel Burnup versus Initial Enrichment~~ Minimum Number of IFBA Rods versus Initial Enrichment	3.8-3
Region 2-1 Limiting Fuel Burnup versus Initial Enrichment	3.8-4
Region 2-2 Limiting Fuel Burnup versus Initial Enrichment	3.8-5
Required Hot Shutdown Margin versus Reactor Coolant Boron Concentration	3.10-1
Vessel Leak Test Limitations	4.3-1
Map Defining Unrestricted Areas for Radioactive Gaseous and Liquid Effluents	5.1-1

not be moved on or above El. 95' in the Fuel Storage Building. Additionally, loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool shall not be moved over spent fuel in the spent fuel pit. The weight of installed crane systems shall not be considered part of these loads.

2. The spent fuel storage pit water level shall be maintained at an elevation of at least 93'2". In the event the level decreases below this value, all movement of fuel assemblies in the spent fuel pool storage pit and crane operations with loads over spent fuel in the spent fuel pit shall cease and water level shall be restored to within its limit within 4 hours.

D. The following conditions are applicable to the spent fuel pit anytime it contains fuel:

1. ~~The spent fuel storage racks are categorized as either Region I or Region II as specified in Figure 3.8-2. Fuel assemblies to be stored in the spent fuel storage racks are categorized as either Category A, B or C based on burnup and enrichment limits as specified in Figure 3.8-3. The storage of Category A fuel assemblies within the spent fuel storage racks is unrestricted. Category B fuel assemblies shall only be stored in Region I or in a Region II spent fuel rack cell with one cell wall adjacent to a non-fuel area (a non-fuel area is the cask area or the area on the outside of a rack next to a wall). Category C fuel assemblies shall be stored only in Region I. The one exception to this shall be fuel assembly F-65 which shall be stored in Region I or in a Region II spent fuel rack cell with two cell walls adjacent to non-fuel areas.~~

Insert A, new 3.8.D.1 first paragraph from next page.

In the event any fuel assembly is found to be stored in a configuration other than specified, immediate action shall be initiated to:

a. Verify the spent fuel storage pit boron concentration meets the requirements of Specification 3.8.D.2, and

b. Return the stored fuel assembly to the specified configuration.

2. At all times the spent fuel storage pit boron concentration shall be at least ~~1500~~ 2000 ppm. With the boron concentration less than this value, all fuel movement within the spent fuel storage pit shall cease and immediate action shall be initiated to restore the boron concentration to at least the minimum specified. The required boron concentration shall be verified by chemical analysis at the frequency specified on Table 4.1-2.

3. During operations described in Specification 3.8.B, the spent fuel storage pit

Insert A – New 3.8.D.1 first paragraph

The spent fuel racks are categorized as either Region 1-1, 1-2, 2-1, or 2-2 as specified in Figure 3.8-1. Fuel assemblies to be stored in the spent fuel storage racks are qualified based on burnup, enrichment, and cooling time as specified in Figures 3.8-2 through 3.8-5.

a. Storage of fuel in Region 1-1 is restricted to assemblies that meet the burnup criteria in Figure 3.8-2.
b. Storage of fuel in Region 1-2 is restricted to assemblies with enrichments ≤ 4.5 weight percent ($^{w}/o$) or < 5.0 $^{w}/o$ with a minimum number of Integral Fuel Burnable Absorbers (IFBA) rods as specified in Figure 3.8-3. Fuel that meets the criteria for Region 1-2 may be stored in Region 1-1 in a checkerboard (1 out of 2 cells with every other cell left vacant) loading configuration.
c. Storage of fuel in Region 2-1 is restricted to assemblies that meet the burnup criteria in Figure 3.8-4.
d. Storage of fuel in Region 2-2 is restricted to assemblies that meet the burnup criteria in Figure 3.8-5. Fuel that meets the criteria for Region 2-1 may be stored in Region 2-2 in "peripheral" cells. As shown in figure 3.8-1, the peripheral cells are located along the west wall and are separated by at least 3 cells.

The requirement for the fuel storage building charcoal filtration system to be operating when spent fuel movement is being made provides added assurance that the offsite doses will be within acceptable limits in the event of a fuel-handling accident. The additional month of spent fuel decay time will provide the same assurance that the offsite doses are within acceptable limits and therefore the charcoal filtration system would not be required to be operating.

The spent fuel storage pit water level requirement in Specification 3.8.C.2 provides approximately 24 feet of water above fuel assemblies stored in the spent fuel storage racks.

The fuel enrichment and burnup limits in Specification 3.8.D.1, the partial credit taken for Boraflex panels, and the boron requirements in Specification 3.8.D.2 assure the limits assumed in the spent fuel storage safety analysis will not be exceeded. The analysis (Ref. 2) takes credit for the amount of Boraflex predicted to be available through 2006.

The requirement that at least one RHR pump and heat exchanger be in operation ensures that sufficient cooling capacity is available to maintain reactor coolant temperature below 140°F, and sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR pumps and heat exchangers operable when there is less than 23 feet of water above the vessel flange ensures that a single failure will not result in a complete loss of residual heat removal capability. With the head removed and at least 23 feet of water above the flange, a large heat sink is available for core cooling, thus allowing adequate time to initiate actions to cool the core in the event of a single failure.

References

(1) FSAR Section 9.5.2

(2) Northeast Technology Corporation Report NET-173-01, "Criticality Analysis for Soluble Boron and Burnup Credit in the Con Edison Indian Point Unit No. 2 Spent Fuel Storage Racks."

~~DELETED~~

~~Figure 3.8-1~~

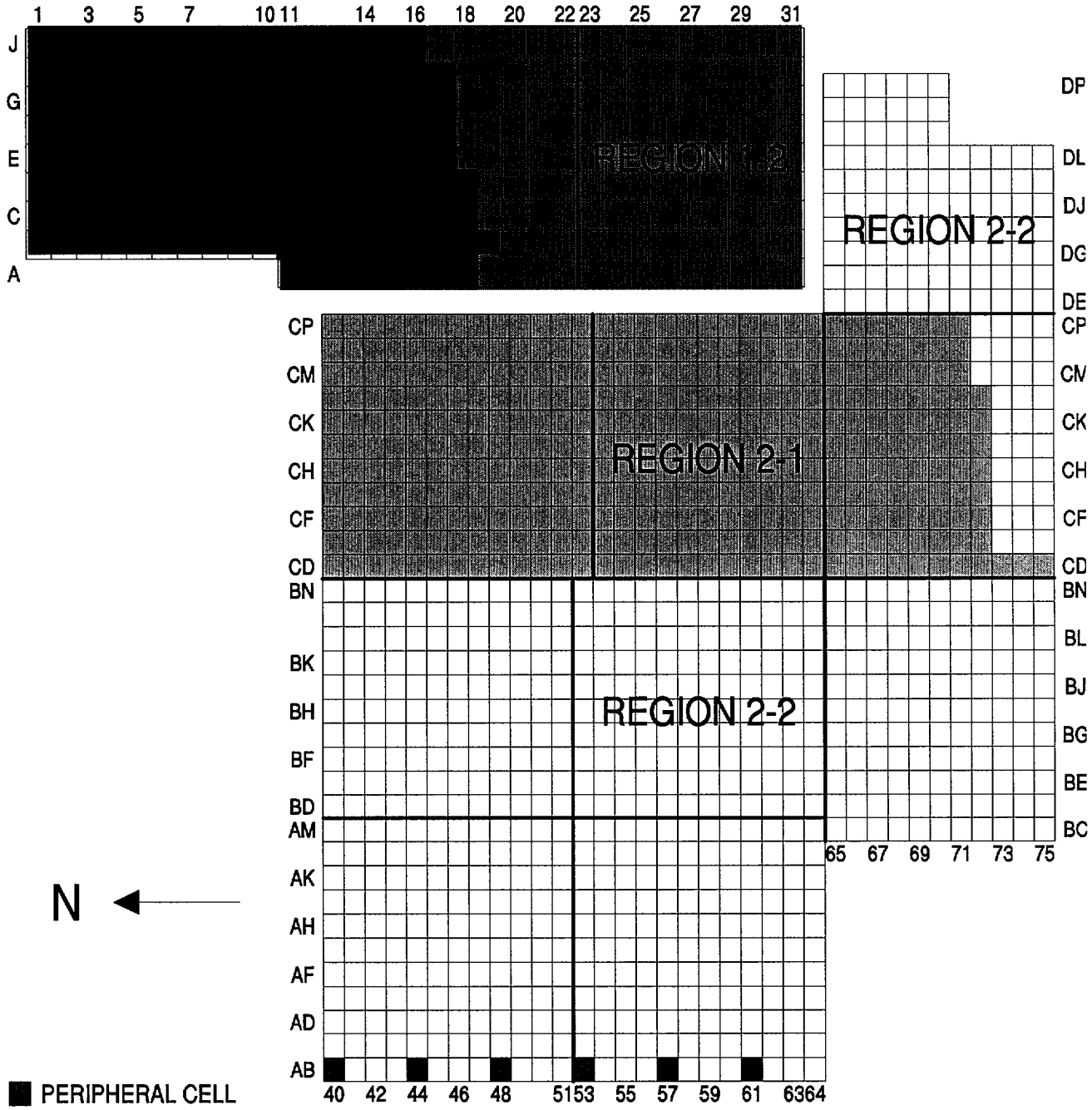
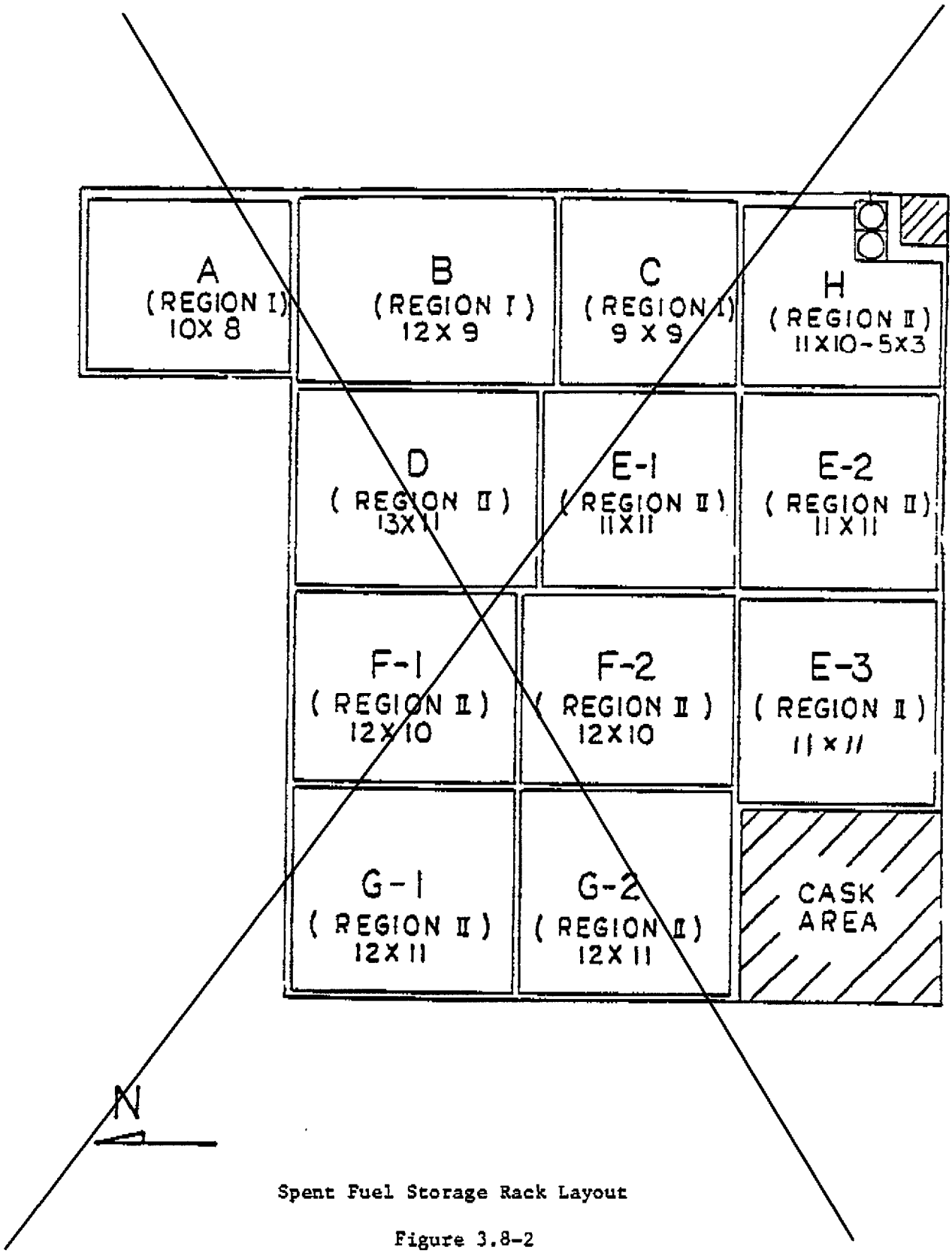


Figure 3.8-1
Spent Fuel Storage Rack Layout



Spent Fuel Storage Rack Layout

Figure 3.8-2

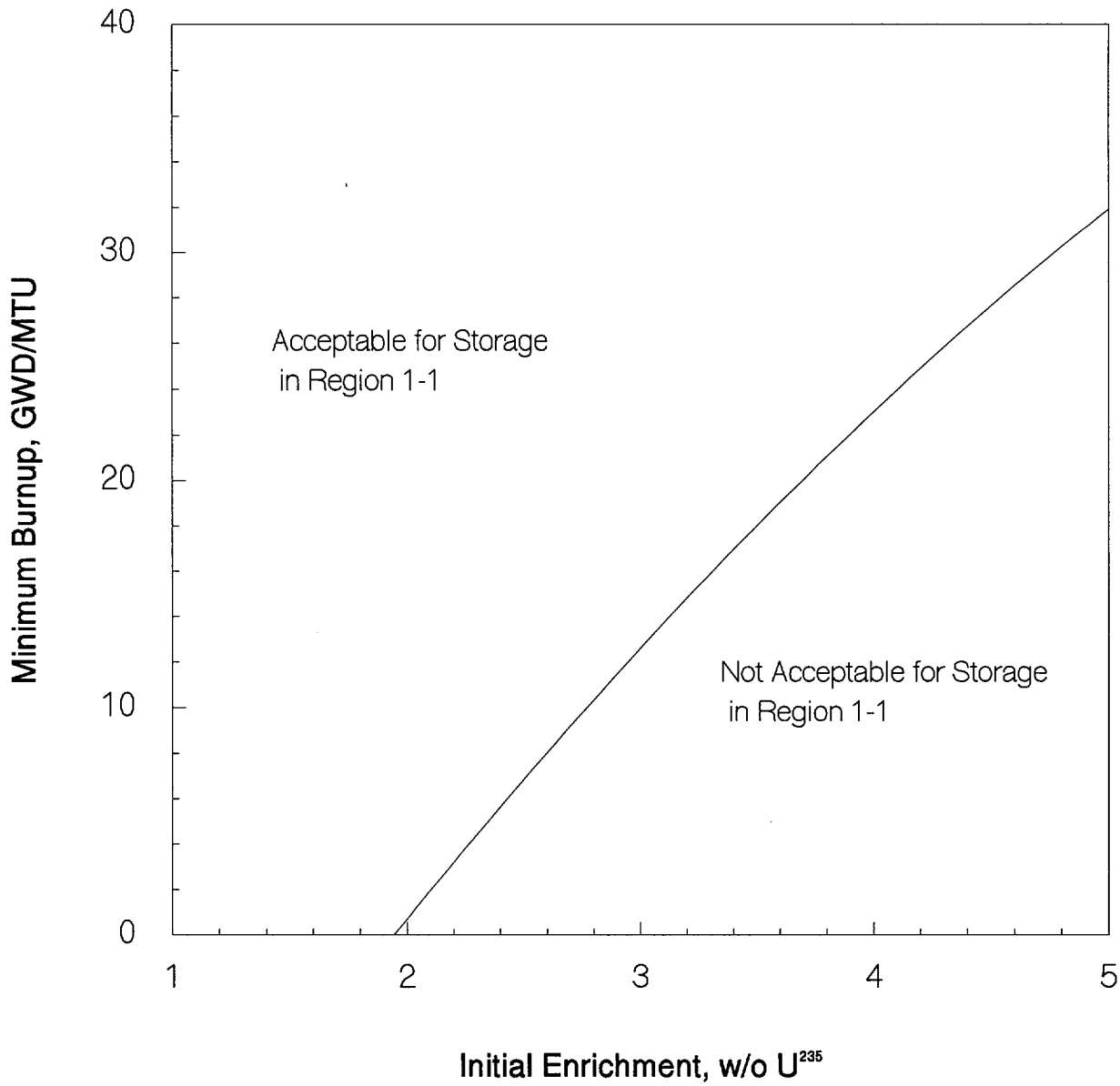
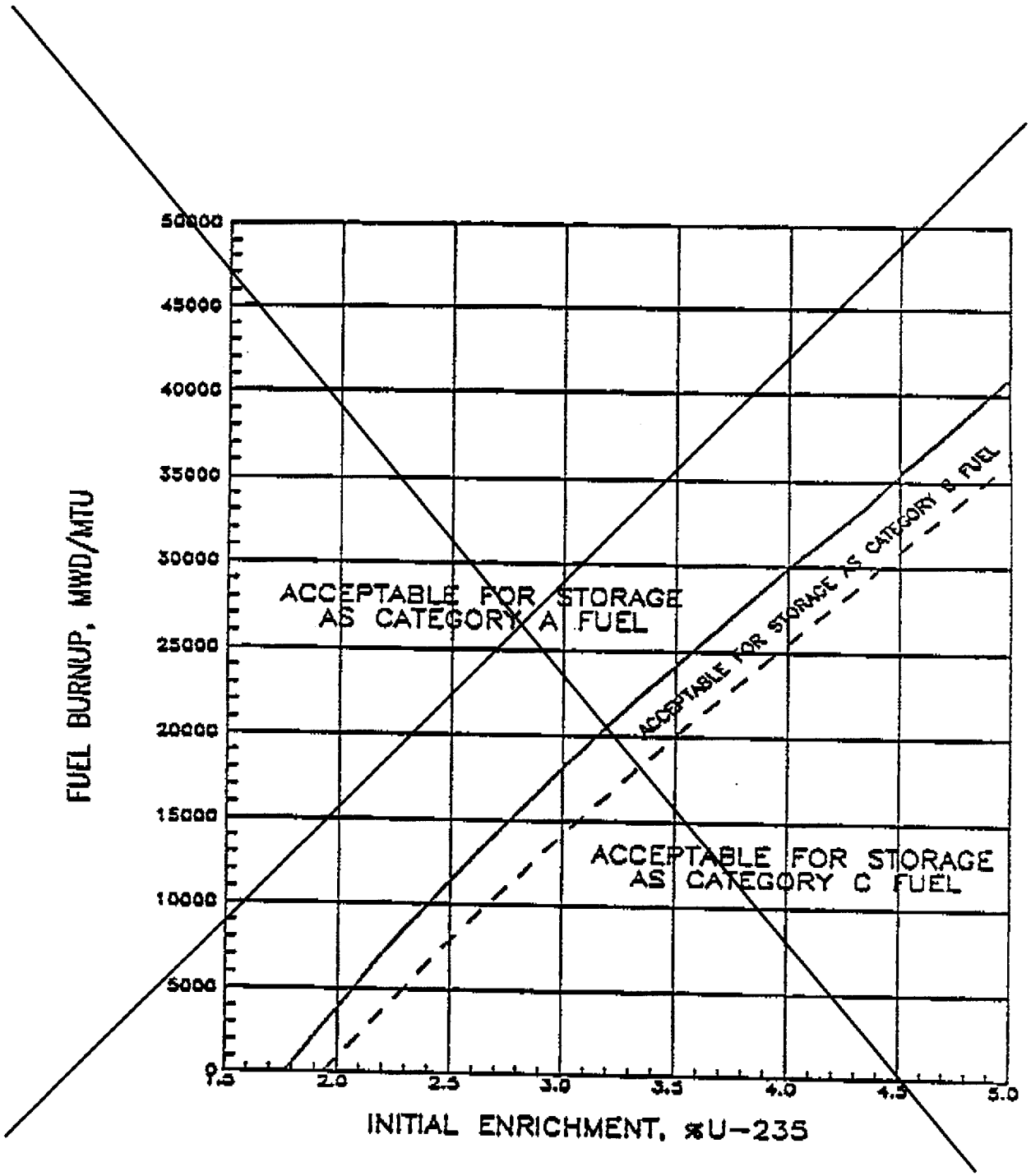


Figure 3.8-2
Region 1-1 - Limiting Burnup versus Initial Enrichment



~~Limiting Fuel Burnup~~
~~Versus~~
~~Initial Enrichment~~

Amendment No. ~~152~~

~~Figure 3.8-3~~

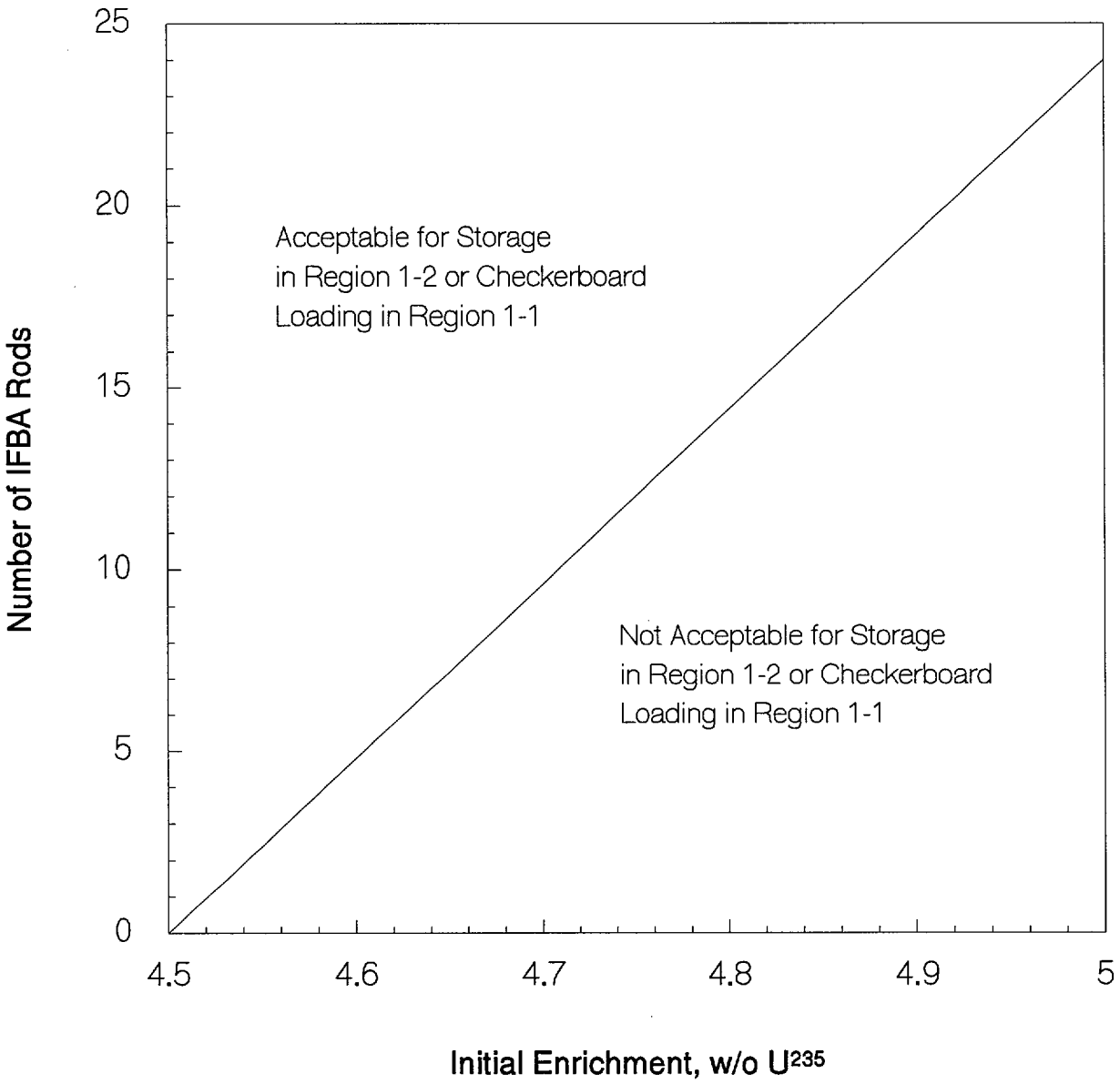


Figure 3.8-3
Region 1-2 - Minimum Number of IFBA Rods versus Initial Enrichment

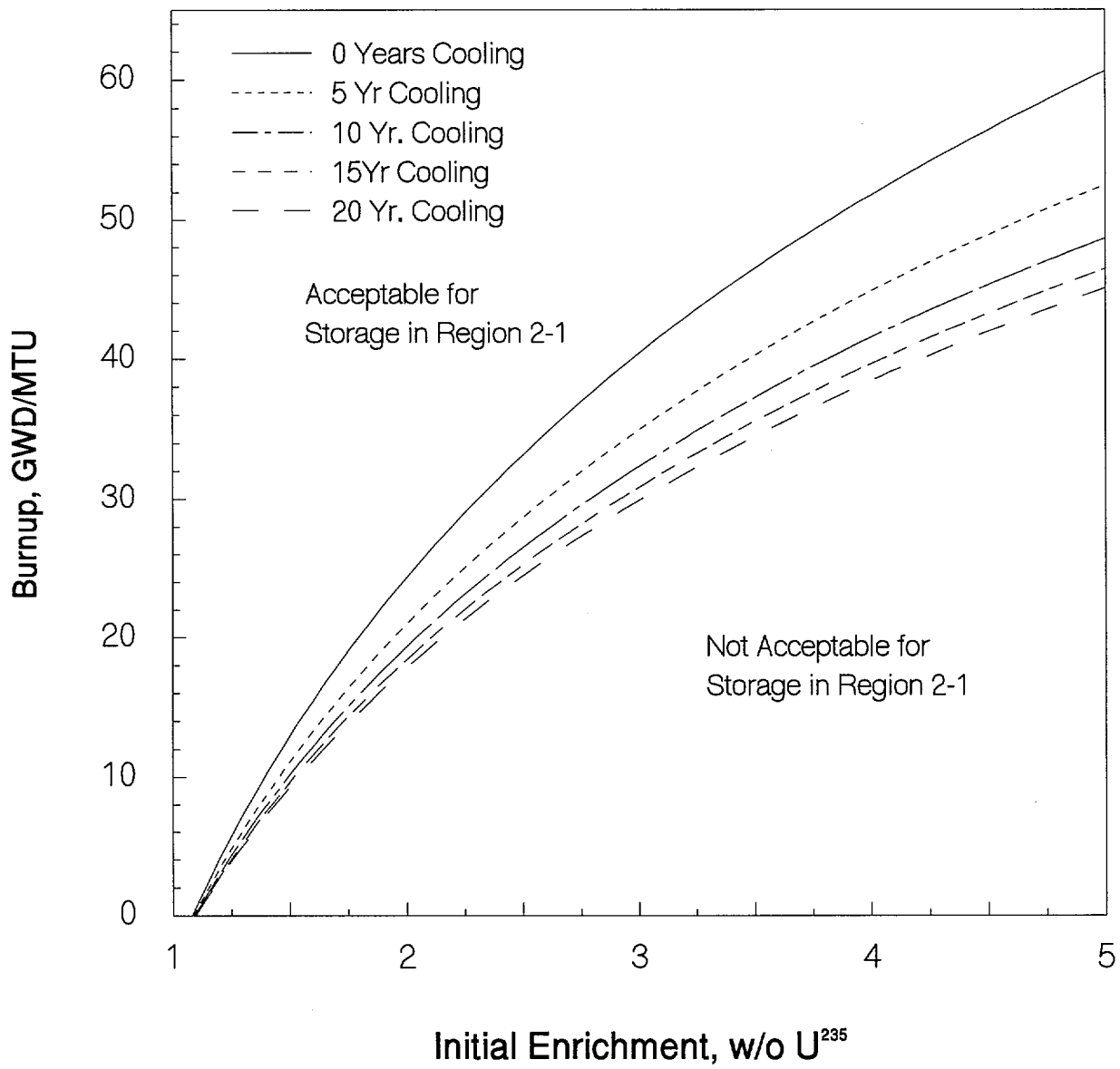


Figure 3.8-4
Region 2-1 - Limiting Burnup versus Initial Enrichment

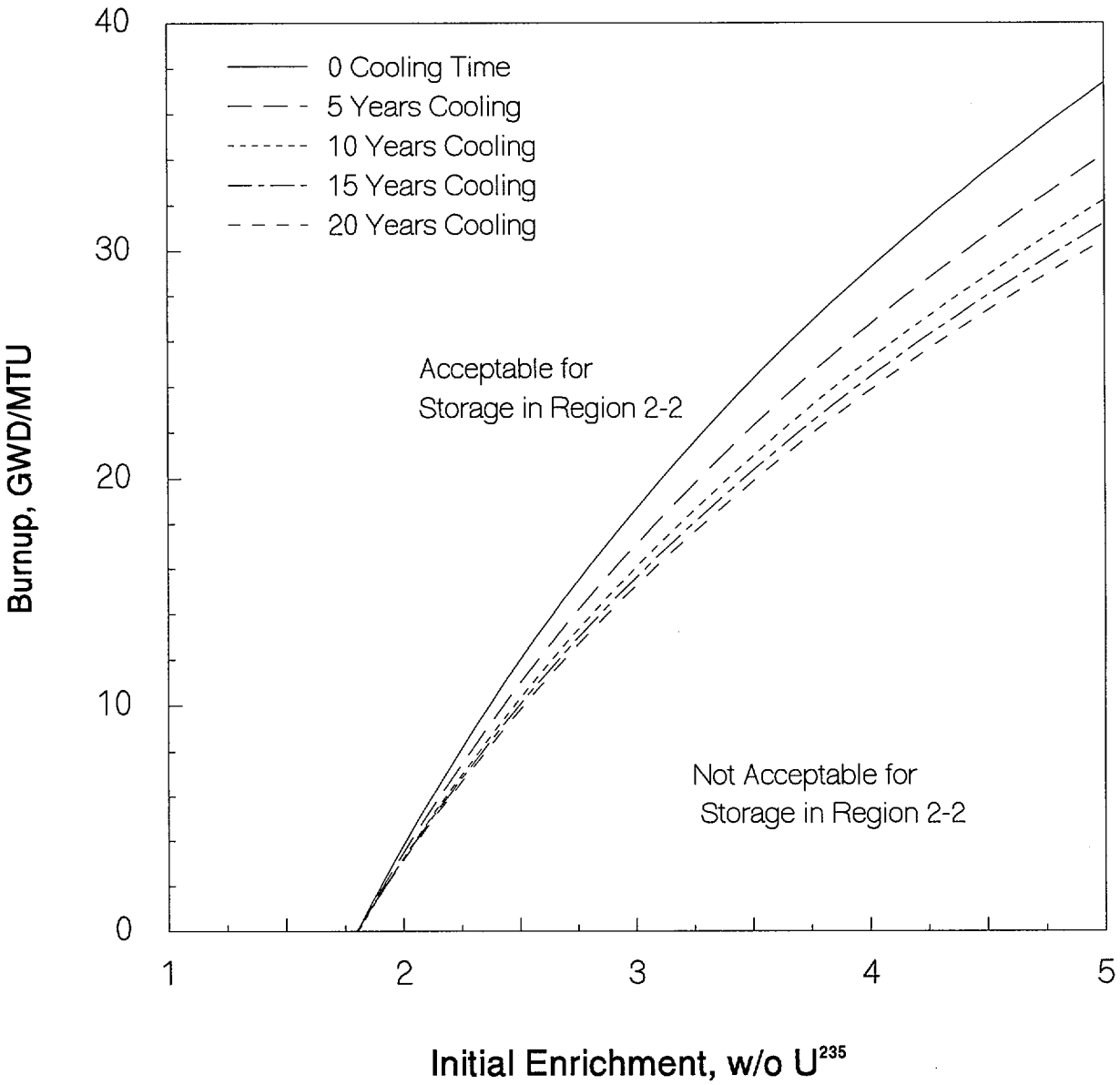


Figure 3.8-5
Region 2-2 - Limiting Burnup versus Initial Enrichment

Table 4.1-2

Frequencies for Sampling Tests

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	Gross Activity (1) Radiochemical(2) $\bar{E}$ Determination Tritium Activity F, Cl & O_2	5 days/week (1) Monthly Semi-annually (3) Weekly (1) Weekly	3 days 45 days 30 weeks 10 days 10 days
2.	Reactor Coolant Boron	Boron Concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Monthly	45 days
4.	Boric Acid Tank	Boron Concentration	Twice/week	5 days
5.	DELETED			
6.	DELETED			
7.	Accumulator	Boron Concentration	Monthly	45 days
8.	Spent Fuel Pit	Boron Concentration	~~Monthly~~ Weekly	~~45 days~~ 10 days
9.	Secondary Coolant	Iodine-131	Weekly (4)	10 days
10.	Containment Iodine Particulate Monitor or Gas Monitor	Iodine-131 and Particulate Activity or Gross Gaseous Activity	Continuous When Above Cold Shutdown(5)	NA*

5.4 FUEL STORAGE

Applicability

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specifications

1. The spent fuel pit structure is designed to withstand the anticipated earthquake loadings as a Class I structure. The spent fuel pit has a stainless steel liner to ensure against loss of water.

2.A. The new fuel storage rack is designed so that it is impossible to insert assemblies in other than an array of vertical fuel assemblies with a sufficient center-to-center distance between assemblies to assure $K_{eff} \leq 0.95$, even if unborated water were used to fill the pit and with fuel assemblies containing a maximum enrichment of 5.0 weight percent U-235, and poisons, if necessary to meet the K_{eff} limit.

2.B. The spent fuel storage racks are designed and their loading maintained within the limits of Technical Specification 3.8.D.1, such that K_{eff} ~~≤ 0.95, even if unborated water were used to fill the pit~~ < 1.0 in unborated water and $K_{eff} \leq 0.95$ with credit for soluble boron and with the fuel assemblies containing a maximum enrichment of 5.0 weight percent U-235 (or equivalent reactivity).

ATTACHMENT 3 TO NL 01-110

NET-173-01
CRITICALITY ANALYSIS FOR SOLUBLE BORON AND BURNUP CREDIT IN THE CON EDISON INDIAN POINT UNIT NO. 2 SPENT FUEL STORAGE RACKS."

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

NET-173-01

CRITICALITY ANALYSIS FOR SOLUBLE BORON AND BURNUP CREDIT IN THE CON EDISON INDIAN POINT UNIT NO. 2 SPENT FUEL STORAGE RACKS

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ABSTRACT

The following report describes the results of analyses performed for Consolidated Edison under Specification No. 00-01, "Specification for Soluble Boron Credits and Other Analyses to Mitigate Boraflex Degradation". In addition to taking credit for soluble boron, the mitigation measures include dividing the spent fuel pool into sub-regions. The results specify minimum requirements for assembly burnup and soluble boron concentration for various spent fuel assembly loading configurations in these sub-regions. The amount of Boraflex absorber panel degradation assumed in each sub-region is based on conservative projections through 2006.

The flux trap type Region 1 racks are divided into two sub-regions. Region 1-1 takes no credit for Boraflex and can safely accommodate assemblies that have been discharged with a minimum burnup, as specified in a burnup versus initial enrichment curve contained in this report. No credit for ^{241}Pu decay is taken in Region 1-1. Alternatively, Region 1-1 can accommodate unirradiated fuel up to 5.0 ^{w}/o ^{235}U, assuming a minimum number of IFBA rods, as specified in this report, using a 1 of 2 checkerboard loading pattern with the alternate cells left vacant. Region 1-2 is assumed to have sustained a 50% loss of Boraflex. This analysis demonstrates that Region 1-2 can accommodate unirradiated fuel up to 5.0 ^{w}/o ^{235}U, assuming a minimum number of IFBA rods, as specified in this report.

The egg crate type Region 2 racks are also divided into two sub-regions. Region 2-1 takes no credit for Boraflex and can safely accommodate assemblies that have been discharged with a minimum burnup and have cooled for a minimum amount of time, as specified by a series of burnup/cooling time versus initial enrichment curves contained in this report. Region 2-2 is assumed to have sustained a 30% loss of Boraflex. This analysis demonstrates that Region 2-2 can accommodate assemblies that have been discharged with a minimum burnup and have cooled for a minimum amount of time, as specified by another series of burnup/cooling time versus initial enrichment curves contained in this report.

Region 1-2 is limiting with respect to minimum soluble boron requirements and would require a minimum of 786 ppm of soluble boron under normal operating conditions. In addition to the normal operating conditions, minimum soluble boron requirements to account for misplaced and dropped fuel bundles have been determined. The worst-case scenario under accident conditions would require 1495 ppm soluble boron to compensate for the associated reactivity increase. Under normal operating conditions, a minimum of 2000 ppm soluble boron will be available in the pool per the Plant Technical Specifications. This is more than sufficient to safely mitigate the complete loss of Boraflex in the spent fuel pool.

1.0 Introduction

In 1990, the Indian Point Unit No. 2 (IP2) 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[1]. The capacity in the pool was increased by decreasing the spacing between adjacent fuel assemblies. This decreased spacing was achieved by using neutron absorbers between rack cells in order to maintain a sufficiently subcritical 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 model of the Indian Point 2 spent fuel pool was developed[2]. The analysis indicated that areas of moderate dissolution of the Boraflex panels had likely occurred. Accordingly, BADGER testing was performed in February 2000 and the results confirmed the predictions of the RACKLIFE computer model[3]. Administrative controls were implemented as a mitigation measure to assure the design basis of $k_{eff} \leq 0.95$ was not exceeded.

One of the features of the RACKLIFE[4,5] program is the ability to perform future predictive calculations to estimate the extent of boron carbide loss and investigate various rack management strategies. Subsequent analyses were completed to assess the extent of boron carbide loss through 2006 when full core offload capability will no longer be available[6,7].

This report describes the criticality analysis of the IP2 SFRs. This analysis takes credit for soluble boron in the spent fuel pool water and parallels analyses performed for other SFRs[8].

In order to mitigate the effects of degraded Boraflex, each of the two regions of storage racks at IP2 have been further divided into two sub-regions. The sub-regions have been divided as follows:

Region 1-1	takes no credit for Boraflex
Region 1-2	takes credit for 50% of the initial Boraflex
Region 2-1	takes no credit for Boraflex
Region 2-2	takes credit for 70% of the initial Boraflex

Within each sub-region, the analyses take credit for burnup depending on the initial ^{235}U enrichment of the fuel assemblies. The unirradiated reload fuel may be stored in either Region 1-1 or Region 1-2. In Region 1-1, reload fuel may be stored in every other storage location (in a so-called "checkerboard" configuration), with empty cells in the alternative locations. In Region 1-2, unirradiated reload fuel may be stored in every storage location.

In addition to burnup and soluble boron credits, reactivity credit is also taken for Integral Fuel Burnable Absorbers (IFBAs) in the analysis of the Region 1-1 and 1-2 SFRs. As noted above, these sub-regions will be used to store the unirradiated reload fuel prior to refueling operations.

The principal design criteria applied to the IP2 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 in all sub-regions is 786 ppm for normal conditions and 1495 ppm for accident conditions.

2.0 Descriptions of the IP2 Fuel Pool, Storage Racks, and Fuel

2.1 Pool Configuration

The IP2 spent fuel pool was re-racked in 1990 to increase the spent fuel storage capacity. The higher density SFRs were designed and fabricated by Holtec International[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 layout of the rack modules in the IP2 fuel pool is shown in Figure 2-1. The total capacity of the pool is 1,374 storage cells.

2.2 Region 1 Rack Cell Design Features

The details of this flux trap rack design are illustrated in Figure 2-2. The basic storage cell for the Region 1 racks consists of four primary elements: 1) the fuel boxes, 2) the Boraflex panels, 3) cover plates to retain the Boraflex, and 4) a series of spacer plates in the axial direction used to separate the fuel boxes in order to form a flux trap between the Boraflex panels.

The fuel boxes are formed from 0.075" thick sheets of type 304 stainless steel. They are 169" long, 8.75" square inside, and are welded to the rack module base plate. The Boraflex panels are nominally 0.1022" thick, 7.5" wide, and 144" long in the Region 1 racks. The neutron absorber boron-10 nominal areal density is 0.0324 g/cm^2. The cover plates are 0.0235" 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 that contains the Boraflex. This cavity is 0.112" deep measuring perpendicular to the cell wall. Five type 304 stainless steel spacer plates 8" high and 0.075" thick are spaced 47.25" apart axially to connect the fuel boxes. They are 1.645" wide in the north-south (N-S) direction, and 1.865" wide in the east-west (E-W) direction. This makes the assembly-to-assembly pitch 10.545" in the N-S direction, and 10.765" in the E-W direction, so the Region 1 racks are diagonally non-symmetric.

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), and Vantage+ design assemblies (with Zircaloy spacer grids, ZIRLO guide tubes, and ZIRLO clad). (It is noted that the top and bottom spacer grids of all designs are Inconel.) The Vantage+ fuel pellet diameters are identical to the OFA, LOPAR, and HIPAR rods, but the fuel density is slightly higher, making this a conservatively higher reactivity assembly to analyze. Vantage+ assemblies, though, possess 6" low enrichment axial blankets at each end, whereas OFA, LOPAR, and HIPAR assemblies are uniformly enriched. In addition, the Vantage+ clad, ZIRLO, has a slightly higher absorption cross section than Zircaloy. Sensitivity analyses of all three assembly types have shown that when the effects of leakage are incorporated, as in the current analysis, the differences in k_{eff} between all three assembly types are negligible. The primary geometric difference between the three 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 (though again by a negligible amount). Thus, a maximum reactivity OFA bundle (with its smaller guide tube and Zircaloy clad) with a maximum fuel density appropriate for the Vantage+ fuel was used for the analyses.

Table 2-1: 15x15 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	5.00

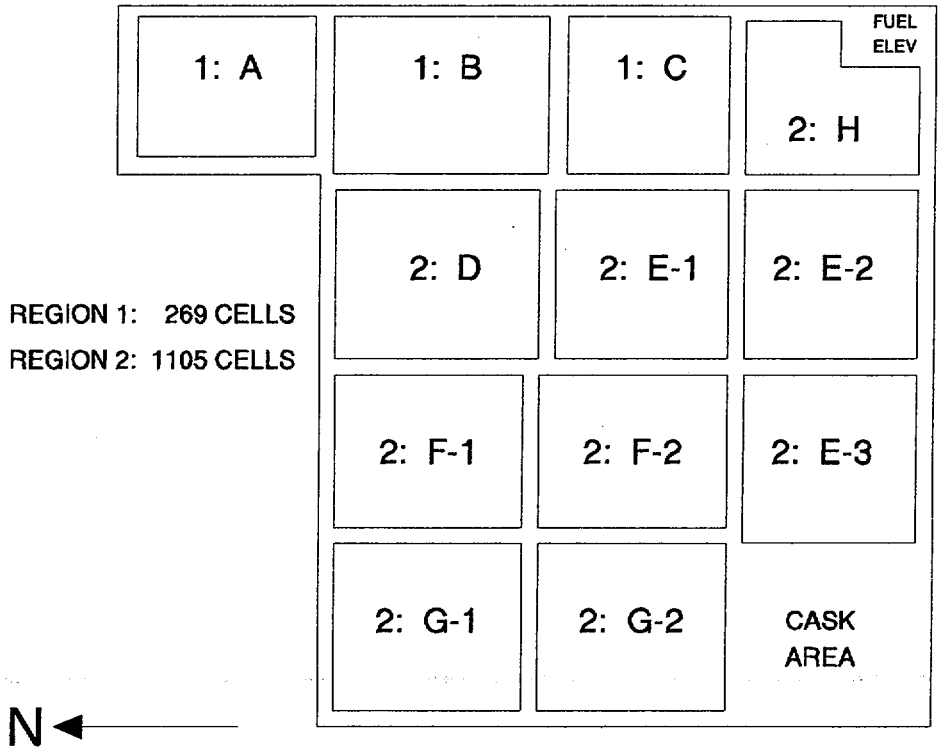

Figure 2-1: Indian Point 2 Spent Fuel Pool Layout with Module IDs

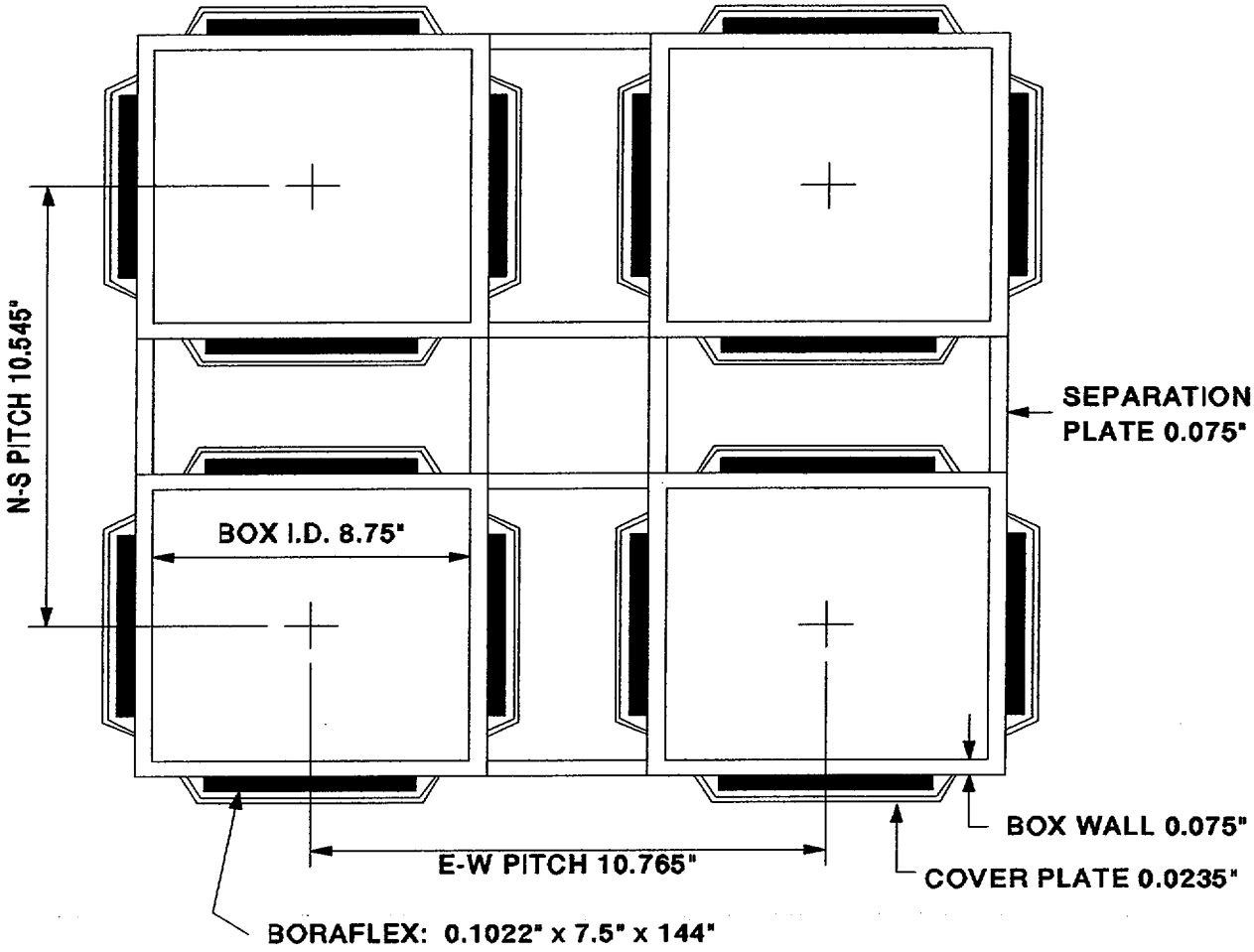

Figure 2-2: Indian Point 2 Region 1 Storage Cells

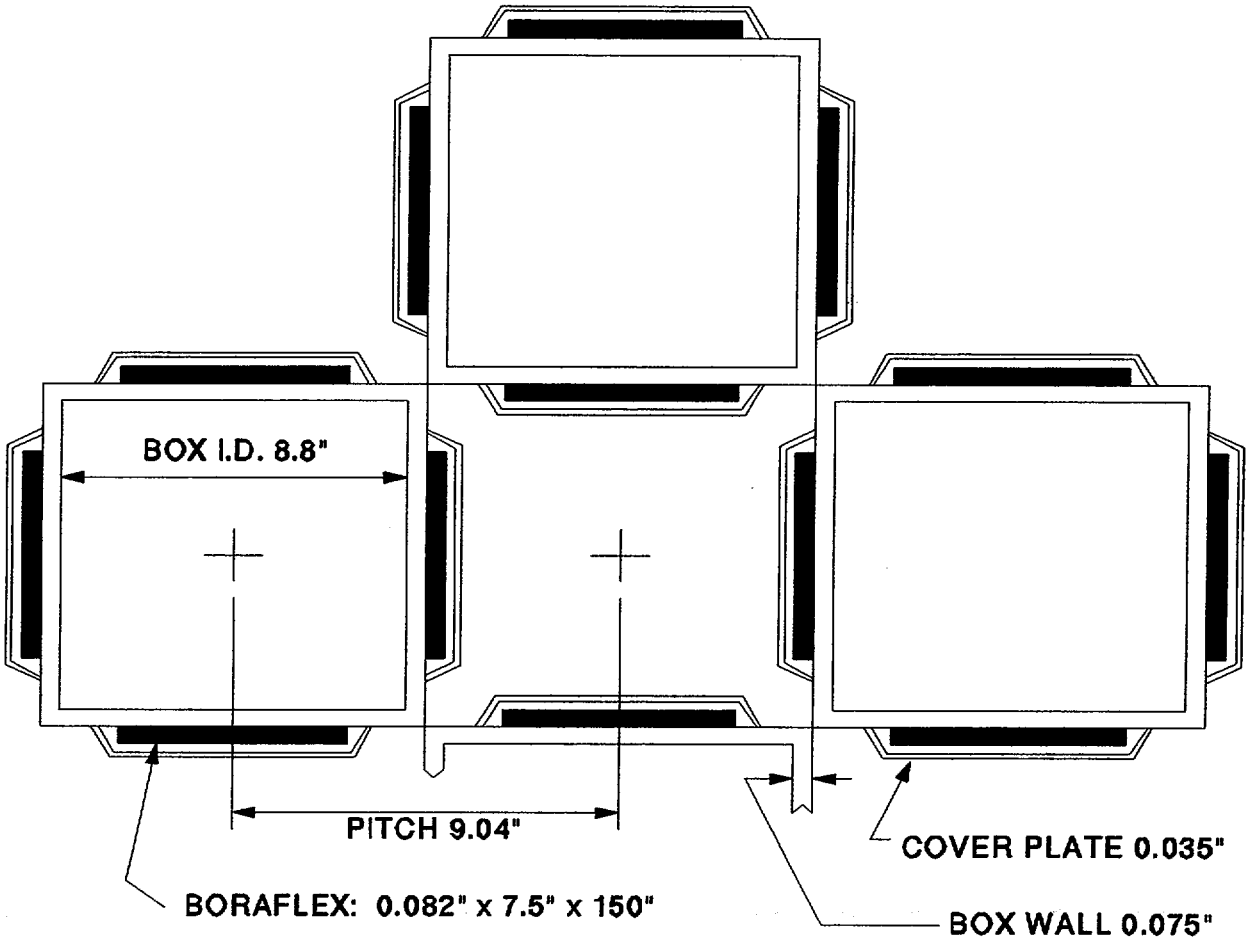

Figure 2-3: Indian Point 2 Region 2 Storage Cells

3.0 Methods and Computer Codes

3.1 Soluble Boron Credits

The methodology used to take credit for soluble boron and burnup parallels analyses performed by the industry when credit is taken for soluble boron[8]. 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, CASMO-4)
- Manufacturing tolerances specific to the IP2 SFRs
- Burnup uncertainty calculations
- Determination of reactivity effect of Boraflex degradation

The differences are a result of the application of the method to the specific fuel assembly types being stored in the Indian Point 2 fuel storage racks, and the specific distribution of Boraflex degradation in the racks.

The basic approach for taking credit for soluble boron and burnup is as follows:

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

2) Once the reference k_{eff} is determined, iteratively determine the minimum soluble boron concentration such that k_{eff} is ≤ 0.95.

3) Employing the technique of reactivity equivalencing, determine through depletion analyses the minimum assembly burnup required (as a function of initial enrichment) for each sub-region.

4) Determine the amount of soluble boron iteratively to account for uncertainties due to burnup, IFBAs, etc.

3.2 Computer Codes

This analysis utilizes the stochastic Monte Carlo code KENO V.a[9] and the deterministic code CASMO-4[10] to compute the reactivity effects due to degraded Boraflex. The CASMO code yields a deterministic solution to the neutron transport equation, which is useful for computing reactivity changes precisely. The stochastic nature of the Monte Carlo solution in KENO means that statistical tolerance factors at 95% probability with 95% confidence must be applied to the solution. On the other hand, CASMO is limited to two-

dimensional (axially uniform) single cell (infinitely reflected) models, while KENO has a very robust three-dimensional modeling capability. Thus, KENO is used when axial effects are important (e.g., axially distributed gaps), or when lateral non-uniformities are present (e.g., checkerboard loading).

KENO V.a is a module in SCALE 4.3, 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 4.3.

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[11-14]. The results of this validation and verification are included in this report as Appendix A[15]. 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[16] was used. The resulting bias in the SCALE codes was calculated to be -0.0087± 0.0042. 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 -0.0113 ± 0.0024.

As noted above, all KENO results require that a one-sided 95% probability / 95% confidence statistical tolerance factor be applied to the computed eigenvalue. In all KENO runs, typically 3000 generations (after skipping between 50 and 400 for source distribution convergence) with between 2000 and 30,000 neutrons per generation were simulated for a total of between 6 million and 90 million neutrons tracked. This typically resulted in statistical uncertainties in k_{eff} of $\sigma < 0.0004$ (one standard deviation) and a 95/95 statistical tolerance factor $\kappa \approx 1.7$[17].

4.0 The Reactivity Effects of Boraflex Degradation

4.1 Introduction

This section examines the reactivity effects of Boraflex panel degradation in the IP2 SFRs. Boraflex panel degradation can be divided into three modes, which are characterized by different degradation mechanisms, as described below.

1. **Uniform dissolution**

   As detailed in Section 2.0, the Boraflex panels in the IP2 SFRs are contained in a "panel cavity" created between the outside of the cell wall and the panel wrapper plate. This panel cavity is filled with water that generally surrounds the Boraflex panel. The exchange of fluid between the bulk pool and the panel cavity (measured by the "escape coefficient") results in a flow across the surfaces of the Boraflex panel. This can lead to a relatively uniform dissolution of the amorphous silica from Boraflex panel surfaces and consequent loss of absorber.

   This mode of degradation increases the transmission of neutrons between assemblies in the spent fuel racks by decreasing the amount of intervening absorber. However, the remaining absorber still interposes between assemblies.

2. **Shrinkage, including gaps**

   Radiation induces crosslinking of the polymer matrix of Boraflex. This causes the material to shrink, reducing the volume of a Boraflex panel. While shrinkage reduces the volume of an interposing panel, shrinkage does not reduce the mass of interposing absorber – that is, the material undergoes densification as it shrinks.

   Width and end shrinkage can "uncover" the active fuel, allowing direct neutron transport between assemblies without any intervening absorber. If a Boraflex panel is not allowed to shrink uniformly (e.g., it is mechanically restrained), gaps will develop. This can lead to direct neutron transport between the centers of assembly faces.

3. **Local dissolution**

   The dissolution described as mode 1, above, is generally uniform. However, local non-uniformities in the panel, panel cavity, and cavity inlet/outlet geometry can accentuate dissolution locally. For example, a gap in a panel locally increases the cavity volume, which locally reduces the effects of wall

friction on flow. This increases the local flow rate that causes degradation. As another example, a bend, bow, or crease in a cover plate can increase the orifice, allowing increased flow into or out of the panel cavity, thereby increasing local degradation. These local effects can exhibit a positive feedback: they accelerate the local dissolution of Boraflex, which increases the local cavity volume. This in turn decreases wall friction losses, increasing local flow rates, further accelerating local Boraflex dissolution.

As suggested in the discussion for each mode of dissolution, each mode will affect the spent fuel pool reactivity differently. These synergistic reactivity effects may be strongly non-linear. Criticality safety calculations using highly bounding assumptions, (e.g., very large gaps all at the assembly mid-plane, complete dissolution of the Boraflex, etc.) lead to reactivity increases far in excess of the actual reactivity state of the spent fuel pool. On the other hand, the non-linear synergy necessitates a robust analysis of the degradation, in order to conservatively take some credit for the Boraflex that remains in the racks. This section of the report outlines a methodology for such a robust analysis.

4.2 Background

The methodology outline will often refer to References 6 and 7, which characterize the state of the Boraflex panels in the IP2 SFRs through the end of calendar year 2006. Reference 6 used the EPRI RACKLIFE code[4,5] to simulate SFP operations at IP2 through the end of 2006 and thereby predict the absorbed dose to and B_4C loss from the IP2 SFP rack Boraflex panels. The simulations rely on a number of very conservative assumptions, which include the following:

- instantaneous reactor shutdown at each end-of-cycle from 100%(3071.4 Mwth through Cycle 15 and 3216.5MWth thereafter) [6] of rated power;
- instantaneous reactor de-fueling at 168 hours following shutdown[6];
- placement of "95th percentile" assemblies (in fuel loading, enrichment, burnup, and power sharing) in every potential discharge location of the pool (not just in the currently available cells, and not just the number of cells required to accommodate the discharge);
- a 45 day residence for assemblies returning to the reactor; and
- a geometric increase in the rate of exchange of fluid between the panel cavity and the bulk pool (the "escape coefficient") over time.

If some of these assumptions prove to be invalid (e.g., an extended outage that precludes returning the spent fuel to the reactor for more than 45 days), it is expected that the

RACKLIFE model can be updated to reflect actual operating conditions and will show that the projections remain conservative.

The results of a BADGER test campaign at IP2[3] were used to characterize the state of the IP2 SFRs Boraflex panels at the time of testing. It further used the RACKLIFE projections[6] (discussed above) to very conservatively project the state of the panels to the end of 2006. These projections led to a classification of the rack cells into various sub-regions. Some sub-regions – "high-loss" sub-regions – were projected to have such high loss in 2006 that a conservative credit for Boraflex would have been difficult to justify. Other sub-regions – "low-loss" sub-regions – were projected to have substantially less loss so that a conservative credit for Boraflex is justified.

Models of the panels in these low-loss sub-regions were developed in Appendices A through D of Reference 7. Further, Appendix E (for the Region 1 racks) and Appendix G (for the Region 2 racks) in Reference 7 present algorithms based on the BADGER data for simulating a Boraflex panel with all of the randomly distributed characteristics of a degraded panel. The input to the algorithms is the panel absorbed dose and B_4C loss predicted by RACKLIFE. The algorithms are based on random sampling from probability distributions developed from the observed BADGER data. The use of normal and uniform random numbers in the algorithms account for the variance observed between RACKLIFE predictions and BADGER observations and the random nature of local dissolution effects.

The Boraflex panel models developed in Reference 7 consist of an array of rectangular blocks: four blocks across a panel to match the four detectors in BADGER, and each block two inches high to match the two-inch "window" in front of the BADGER detectors. The blocks are nominally as thick as a minimum certified panel of Boraflex. Each panel of Boraflex in the IP2 SFRs that was measured by BADGER was characterized using this system of blocks. Figure 4-1 is an example of a panel model. In Figure 4-1, the column heading "Elev" refers to the axial elevation of each block center. (The panel shown represents a 144 inch Region 1 panel; note that the panel is displayed bottom to top.) The columns are numbered to correspond to the four BADGER detectors.

Integer values in the Figure 4-1 panel model represent an amount of gap in a cell in 1/3^{rds} of an inch. Thus the row of "3"s on a red background indicates a one-inch gap at an elevation of five inches. Another row of "3"s on a green background at 143 inches of elevation is distinguished because it represents end shrinkage as opposed to a gap. Cells in the panel model that are not colored are at a specified level of uniform loss. The values on blue backgrounds represent areas of local dissolution, quantified by the percent loss from the uniform loss condition. Some of the dissolution occurs around the gap, some near the end of the panel, and some independent of any other features of the panel. Dissolution that occurs around a feature is assumed to extend into the feature. For example, the 60% loss measured by detector 2 (column 2) at 141 inches is assumed to persist in the column 2 cell at 143 inches. In reality, BADGER would detect the additional loss if it was there, but this accounts for any uncertainty in an analyst's interpretation of

how to allocate the loss. In the case of the gap at 5 inches a loss of 50% is assumed under detectors 3 and 4 since this is (conservatively) the largest loss proximal to the gap.

In applying the panel models to the state of the IP2 SFRs in 2006, the degree of conservatism used is best illustrated by the following examples.

Example 1: Loss Equivalence

The BADGER campaign at IP2 in February 2000 measured the state of the IP2 SFRs Boraflex panels at that time. The RACKLIFE code was used to identify which panels in the IP2 SFRs had the highest absorbed dose and/or the highest predicted B_4C loss. Measurements were performed on a spectrum of dose and loss (in order to observe and quantify any trends with dose and loss), but with a strong bias toward the "worst" panels. Therefore, the panels that BADGER measured are typical of the worst panels in the pool. The highest loss panel observed in the Region 1 racks was 17.9% below its expected nominal B_4C loading, and the highest loss panel observed in the Region 2 racks was 21.7% below its expected nominal B_4C loading.

By the end of 2006, RACKLIFE predicts that the average loss in the low-loss sub-regions of the Region 2 racks is 7.9% $\pm$ 3.0%, with a maximum loss of 22.3%. These loses are comparable to or bounded by what BADGER measured. Thus, in predicting what a 10% loss panel (for example) will look like in 2006, it is reasonable to assume it looks like an equivalent 10% loss panel that BADGER measured in 2000. If a 10% loss panel is not available, the next higher loss panel measured can be conservatively used. In this manner, projected panels in 2006 are conservatively loss-equivalenced to panels measured by BADGER in 2000.

Example 2: Loss Extrapolation

By the end of 2006 RACKLIFE predicts that the average loss in the low-loss sub-regions of the Region 2 racks is 21.8% $\pm$ 3.5%, with a maximum loss of 26.8%. This exceeds the maximum loss observed by BADGER of 17.9% by a factor of 1.5. The question is, what will a 26.8% loss panel look like in 2006, given what a 17.9% loss panel looks like in 2000: how can the loss be extrapolated? Of the three modes of degradation described in Section 4.1, the first two, uniform dissolution and shrinkage, are easy to conservatively project with confidence and precision. Their physical mechanisms are well understood and bounding models can be formulated. The third mode, local dissolution, however, is highly random in nature and is not easily bounded.

For example, consider a typical local dissolution feature: a "scallop" in the side of the panel where higher levels of loss are observed. As illustrated below, suppose this takes the form of two 2" high by 1.75" wide rectangular cells along the left edge of the panel with 30% more loss than the uniform loss of the bulk panel. (The rectangular cells bound the actual size and shape of the scallop.)

The question is, more specifically now, what will this local dissolution feature look like in a panel that has undergone 1.5 times as much dissolution? Three distinct degradation scenarios can be considered: 1) the scallop increases in size by a factor of 1.5 (to three cells instead of two); 2) the scallop "deepens" by a factor of 1.5 (from 30% loss to 45% loss); or 3) the scallop remains the same and another one-cell scallop with 30% loss develops somewhere else on the panel. The truth is likely a randomly weighted mixture of all three modes. To select a bounding degradation scenario is virtually impossible, since the reactivity effects of each scenario will depend on the elevation of the scallop, its proximity to other local dissolution features and gaps, etc. The conservative approach used was to assume all three scenarios occur simultaneously on a cell-by-cell basis. Thus, the extrapolated panel is assumed to look something like the following.

Following the degradation scenarios above: 1) the area of each cell has been increased in size by a factor of 1.5 (which rounds up to two cells since only an integer number of cells are considered in the model); 2) each original cell is deepened by a factor of 1.5; and 3) a new scallop is randomly placed at another location in the panel. While highly conservative, this insures that the reactivity effects of extrapolating local dissolution features are bounded.

Using the panel projections described above, the following methodology was developed for simulating the reactivity effects of Boraflex panel degradation.

4.3 Methodology for Assessing the Reactivity Effects of Boraflex Degradation

The methodology described below was applied to the IP2 Region 1 and Region 2 SFRs independently. For clarity, the description below will generally refer to the racks generically.

The SCALE code package (described in Section 3.2) was used to calculate k_{eff} for the racks. For the base case, the Boraflex was assumed to be at its minimum certified thickness and ^{10}B loading. In addition, a conservatively bounding 4.1% width shrinkage was also applied. This bounding shrinkage is based on both analytical and experimental analyses[2] and has been confirmed by a large number of proprietary laboratory studies and field observations. Recall from Section 4.1 that thickness shrinkage is effectively offset by densification and so need not be accounted for. As described in Section 4.1, the effects of axial shrinkage manifest themselves as both end shrinkage and gapping. Measuring the amount of shrinkage-induced gapping is complicated by the fact that local dissolution can increase the apparent size of a gap. Further, BADGER may miss gaps that are less than 1/3rd inch or smaller. To account for the axial shrinkage with the possibility that some gaps may have been missed, it is conservatively assumed that every panel has an undetected 4.1% axial shrinkage in the form of 1/3rd inch gaps uniformly distributed up the panel. This is in addition to whatever shrinkage effects are apparent from the BADGER scans. The reactivity effect of this assumption is shown in Table 4-1. These assumptions result in a higher than nominal reactivity model, which conservatively increases the reactivity effects of Boraflex loss.

The Boraflex thickness in the base model was then uniformly decreased in 5% increments to observe the reactivity effects of uniform dissolution. The results were used to develop a relationship between uniform thinning and an increase in k_{eff} for reactivity equivalencing between pure uniform thinning and the actual degraded condition of the Boraflex. The results are shown in Table 4-2.

Next, a verified and validated Fortran program was developed to modify the base case, so that every panel in a given array of rack cells could be modeled independently. The algorithms described in Section 4.2 were used to create panel models as described in that section for each panel in the array. For example, Region 1, Module C in the IP2 SFRs is a 9x9 array of cells of the flux trap design with four Boraflex panels per cell (as described in Section 2.2). Thus, a total of $9 \cdot 9 \cdot 4 = 324$ panels are generated by the algorithm according to the dose and loss predicted by RACKLIFE for each panel. These models are incorporated into a KENO model to simulate the module and its (randomly) degraded panels. This case is used to calculate a single estimate of the reactivity effect of Boraflex panel degradation in Module C in 2006.

In executing the case, a total of 30 million neutrons were tracked over 3000 generations. Fifty generations were skipped to ensure convergence of the source distribution. The

large number of neutrons was used to ensure that there was adequate sampling of all of the degradation features of all of the panels in the model. As per standard practice, plots and statistics of the evolution of k_{eff} by generation were inspected and calculated to provide confidence that no sampling instabilities were being encountered.

As described in Section 4.2, the Boraflex panels generated for a model were based on a sequence of random numbers, so that each panel model is a random model with an expected value defined by the BADGER measurements plus a random variance. Consequently, the single estimate case described above could be randomly higher or lower than the actual condition of the panel being modeled. Therefore, a total of 50 independent and identically distributed cases were created using the Fortran program. These cases resulted in a distribution of calculated reactivity effects. The 95th percentile of this reactivity effects distribution, at 95% confidence, can be used to bound the reactivity effects of degraded Boraflex panels in the array of cells being considered. Figure 4-2 shows one example of this distribution as points in a cumulative distribution with the Monte Carlo statistical uncertainty, as shown by the error bars. The line in Figure 4-2 is a cumulative normal distribution with a mean and variance from the fifty samples. In every distribution calculated, the data passed the Anderson-Darling and Cramér-von Mises tests for normality; thus, one-sided normal distribution statistical tolerance factors are valid for calculating bounding 95th percentile eigenvalues at 95% confidence. Figure 4-2 shows that fifty samples are sufficient to bracket the 95th percentile and to look for any potential non-normal behavior in the tails. No non-normal behavior was observed.

4.4 Results

Table 4-3 summarizes the reactivity effects in some of the areas of the IP2 SFRs where credit for Boraflex is being utilized. Modules (or parts of modules) that have bounding distributions of panel loss relative to other modules in the region taking credit for Boraflex control the amount of credit that will be taken. A low-loss module is also shown for each region to show the degree of conservatism in those areas. The RACKLIFE predicted loss for each case, as a uniform thinning loss, is shown in column 4. The RACKLIFE code does not distinguish between uniform loss and local dissolution losses. The reactivity effect in column 5 is the 95th percentile effect at 95% confidence and includes the effects of uniform dissolution, local dissolution, and gaps.

Table 4-2 was used to interpolate the equivalent amount of uniform thinning loss that will yield the same reactivity effect as the 95/95 effect above. The results are shown in column 6. The value of 16.0% for the equivalent loss in the low-loss racks in Region 2 is likely a significant over-estimate of the actual equivalent loss. Most of the panels measured by BADGER in Region 2 had very high predicted and observed losses compared to the losses predicted for the population of Region 2 panels. Thus, in equivalencing observed panel losses with predicted losses, a large amount of conservatism was introduced for the low loss panels.

Column 7 shows the conservative amount of uniform thinning loss that will be assumed in subsequent analyses. The many conservatisms used to arrive at these numbers provides confidence that these losses will bound the state of the IP2 SFRs at the end of 2006.

Table 4-1: Conservative Reactivity Effects of Cracks Undetected by BADGER

Reactivity Effect	Region 1	Region 2
$\Delta k_{eff\text{-}95/95}$	0.00455	0.00502

Table 4-2: Reactivity Effects of Uniform Boraflex Panel Thinning

Thinning Loss [%]	Region 1 Δk_{eff}	Region 2 Δk_{eff}
0.0% (Base)	0.00000	0.00000
5.0%	0.00128	0.00211
10.0%	0.00209	0.00438
15.0%	0.00368	0.00661
20.0%	0.00463	0.00928
25.0%	0.00646	0.01173
30.0%	0.00767	0.01441
35.0%	0.01012	0.01772
40.0%	0.01205	0.02088
45.0%	0.01485	0.02484
50.0%	0.01763	0.02928

Table 4-3: Reactivity Effects of Degraded Panels

Rack Region	Rack Module	Description of Loss	Predicted Loss[1]	Reactivity Effect[2], $\Delta k_{95/95}$	Equivalent Loss[3]	Assumed Loss[4]
1	Part of B	Bounding	20.8% ± 2.5%	0.01439	44.2%	50.0%
1	C	Low	4.4% ± 5.4%	0.00229	10.6%	50.0%
2	E3	Bounding	8.7% ± 5.9%	0.01009	21.6%	30.0%
2	H and Part of E2	Low	3.3% ± 1.7%	0.00714	16.0%	30.0%

[1] This is the average ± 1σ loss predicted by RACKLIFE for the zone being analyzed.

[2] This is the 95th percentile at 95% confidence reactivity effect of the degraded Boraflex panels in the zone being analyzed.

[3] Based on Table 4-2, this amount of uniform thinning will result in the same reactivity effect as shown in the previous column.

[4] This is the conservatively higher amount of Boraflex loss (as uniform thinning) that will be assumed in subsequent analyses.

Elev	1	2	3	4
1				
3			50.0%	50.0%
5	3	3	3	3
7			25.0%	30.0%
9			25.0%	
11				
13				
15				
17				
19				
21				
23				
25				
27				
29				
31				
33				
35				
37	30.0%			
39	45.0%			
41	45.0%			
43	30.0%			
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103				
105				
107				
109				30.0%
111				30.0%
113				
115				
117				
119				
121				
123				
125				
127				
129				
131				
133				
135				
137				
139	70.0%			
141	40.0%	60.0%		
143	3	3	3	3

Figure 4-1. Typical Model of a Region 1 Panel from Reference 7

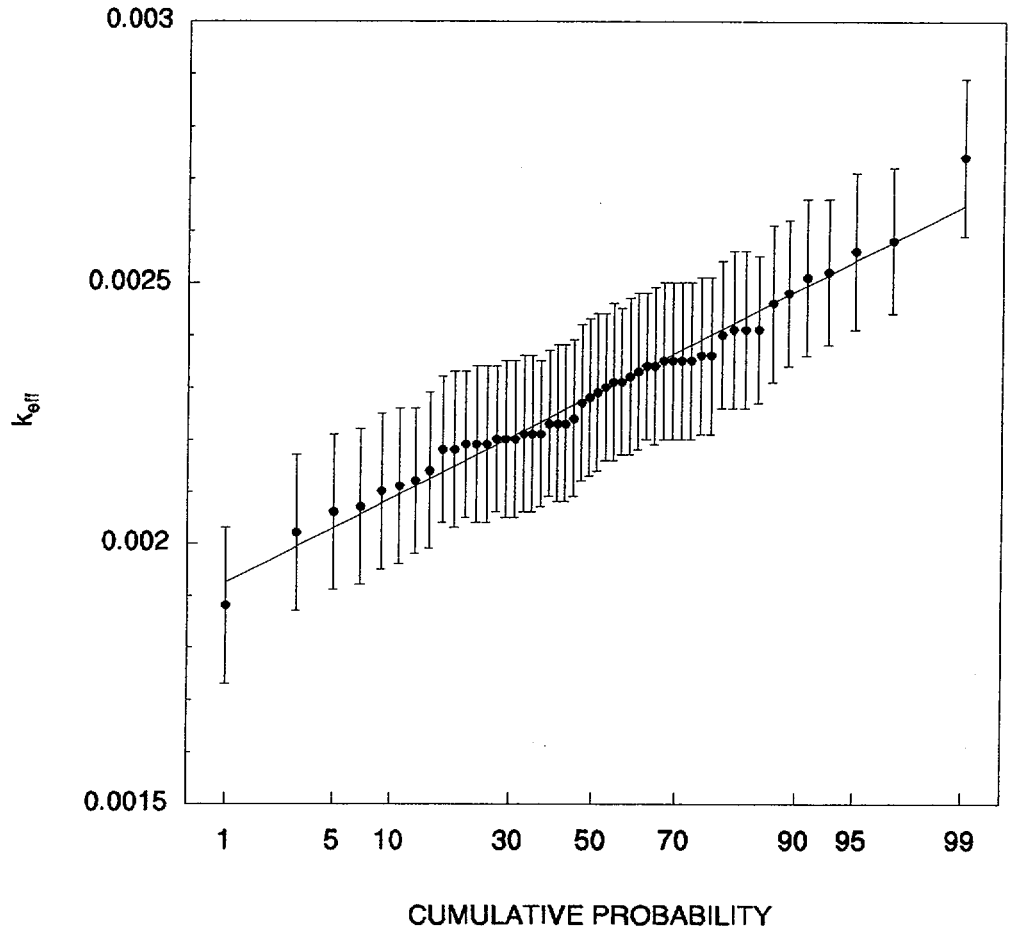

Figure 4-2: Sample Distribution of Panel Degradation Reactivity Effects

5.0 Assumptions, Biases, and Uncertainties for Criticality Analysis

5.1 Reference Analysis

Figure 5-1 shows the layout of the various sub-regions in the IP2 spent fuel pool. In determining the reference k_{eff} for each sub-region, several basic criteria were incorporated into the basic cell model. The reference models consisted of infinite arrays of storage cells in the X-Y direction, each filled with assemblies as specified in Table 2-1. Axial neutron leakage has been included in each region and the end region above the active fuel has been modeled as a water reflector. For regions where Boraflex is credited, the thickness is taken at a fraction of the minimum as-built thickness. All models assume the Boraflex width is at the minimum as-built width, less 4.1% to account for width shrinkage. For the Boron-10 density in the panels, the density is taken as the minimum ^{10}B density (g/cm^3) that corresponds to the minimum certified areal density (i.e., minimum ^{10}B density multiplied by the minimum thickness). No credit is taken for other components in the Boraflex.

In accordance with standard practice[18], the following tolerances and uncertainties were investigated:

- Cell Inner Dimension Tolerance
- Cell Wall Thickness Tolerance
- Cell Pitch Tolerance
- Flux Trap Dimension Tolerance (Region 1 only)
- ^{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 5-1 lists the manufacturing tolerances, as determined via manufacturing specifications, as-built drawings, or benchmark calculations. Column 1 lists the specific

tolerance, while columns 2 and 3 list the values for Regions 1 and 2, respectively. Because the Boraflex is modeled at its minimum certified dimensions, no sensitivity analysis with respect to Boraflex dimension is necessary.

In addition, the following biases were also accounted for:

- Calculation Methodology Bias
- Reactivity Equivalencing Bias
- Potential Bias in Measurements of Boraflex Degradation
- Discrete Absorber Particle Self-Shielding Bias

The first two biases listed are implicitly incorporated into the results reported subsequently in Section 6.0. 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[19]. For the analyses performed here, this bias is only applicable in calculations involving misplaced bundles and the interface between regions.

The last two biases listed above are only applicable to regions that take credit for Boraflex. The potential bias in measurements of Boraflex degradation is from Table 4-1 and is described in Section 4.3. This is a very conservative bound on the reactivity effects of Boraflex degradation that BADGER may have missed. The discrete absorber particle self-shielding bias accounts for the fact that Boraflex is made from discrete boron carbide particles and thus is not a homogeneous distribution of absorber.

5.2 Uncertainties Introduced by Depletion Analyses

5.2.1 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[20]. Measured isotopics for the most important 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, are given. 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 specified[20]. After depletion, an assembly consisting of G9 rods was placed in a fuel rack and the k_{eff} calculated. The assembly and racks were again modeled with isotopes, 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 would extrapolate to +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 isotopes is +0.01 Δk at 60 GWD/MTU.

5.2.2 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 assume a uniform axial power shape. Certain conditions can occur in the reactor (e.g., control rod position, 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 data available indicated that in IP2 Cycles 14 and 15 (current cycle), 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 nodal burnups for 24 axial nodes for Cycles 14 and 15 were evaluated.

The SCALE 4.3 depletion module SAS2H was utilized to determine the depletion dependent isotopics for each of the 24 axial nodes for assembly R08 at the end of Cycles 14 and 15. The generated isotopics were then input into a discrete 24 axial node KENO V.a model of the racks and the k_{eff} calculated. A discrete axial model was created for each of the SFR sub-regions where depleted fuel is to be stored, specifically Regions 1-1, 2-1, and 2-2.

As a reference comparison, similar uniform axial burnup models were created for each SFR sub-region. These models assume that all nodes 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 node explicit model and the uniform axial model were calculated. The results, summarized in Table 5-2 for end-of-cycle (EOC) 14 and 15, are the reactivity effect of non-uniform depletion at 95% probability with 95% confidence.

5.2.3 Removal of Burnable Absorbers

Both the current Vantage+ fuel type and earlier fuel types utilize removable burnable poison assemblies. These include the Wet Annular Burnable Absorbers (WABAs), which may be removed after one cycle of operation. Since the WABAs tend to harden the neutron spectrum, an assembly containing WABAs may not achieve the burnup that assemblies without any WABAs achieve.

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, clean rack condition) were compared as a function of burnup. The assembly with WABAs present was depleted and the WABAs removed prior to placement in the cold clean rack condition. The k_{eff} of this configuration was compared to the same scenario with an assembly that never contained WABAs and the Δk computed as a function of burnup. The maximum difference was +0.00951 Δk. Accordingly, for the present analysis it will be assumed that the maximum reactivity effect due to removal of a burnable absorber will be +0.01 Δk. The amount of soluble boron required to offset +0.01 Δk was then calculated for each region and is described subsequently.

Table 5-1: Tolerances for the IP2 SFRs

Tolerance	Region 1	Region 2
Cell ID	± 0.030	Same
Cell Wall Thickness	± 0.005	Same
Cell Pitch	± 0.035	± 0.040
Flux Trap Width	± 0.016	N/A
Enrichment ($^{w}/o$ ^{235}U)	± 0.05w/o	Same
UO_2 Density	± 2%	Same
Fuel Pellet Dishing	0%-2% (0%-1% each end)	Same
Asymmetric Fuel Position	Offset (relative to Centered)	Same

Table 5-2: Axial Burnup Effect on Reactivity of Assembly R-08 in Core Location H-8

Cycle	Region 1-1	Region 2-1	Region 2-2
EOC-14	0.00261	0.00788	0.00427
EOC-15	0.01345	0.02945	0.01962

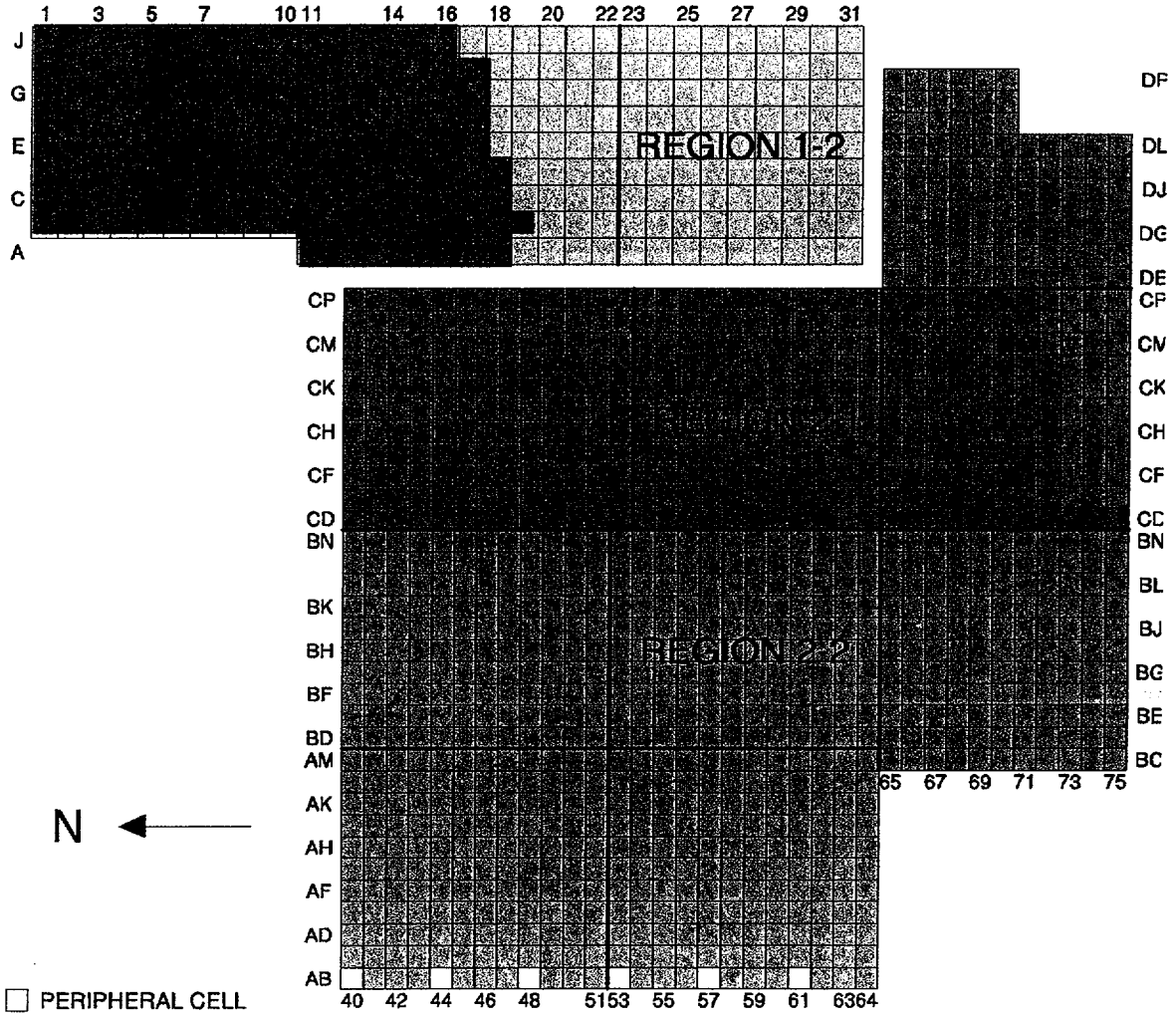

Figure 5-1: Indian Point 2 Spent Fuel Storage Regions.

6.0 Results of the Criticality Analysis

6.1 Region 1-1, 0% Boraflex Credit

6.1.1 Reference Model, Including Tolerances and Uncertainties

Analyses determined that with 100% panel loss, the maximum initial enrichment that can be safely stored in Region 1-1 is 1.95 $^{w}/o$ ^{235}U at zero burnup. A KENO V.a model of an infinite array of fuel assemblies in the X-Y direction with water reflectors above and below the fuel region results in a k_{eff} of 0.96970. This value has been corrected, as appropriate, for both the calculation methodology bias and the reactivity equivalencing bias noted in Section 5.1. This reference model assumes the rack is at a nominal pool temperature of 68°F with water at full density and no soluble boron. Table 6-1 lists the associated reactivity effects of each of the manufacturing tolerances. The statistical combination of uncertainties and tolerances adds an additional 0.01167 Δk to the reference k_{eff}, resulting in a 95% probability at 95% confidence (95/95) upper statistical limit on k_{eff} of 0.98137. This is below the limit of 1.0 without credit for soluble boron.

6.1.2 Soluble Boron Credit

In order to assure that k_{eff} remains below 0.95 with soluble boron, limited soluble boron credit ($\Delta k = 0.05$) is used. This lowers the best estimate 95/95 k_{eff} with boron to $k_{eff} \leq 0.95$. This is achieved iteratively by varying the soluble boron concentration (ppm soluble boron) until the k_{eff} with soluble boron is equal to the k_{eff} with no soluble boron less 0.05 Δk. Iterating on soluble boron concentration results in a minimum required concentration of 163 ppm soluble boron to assure $k_{eff} \leq 0.95$. Table 6-1 lists the associated reactivity effects calculated with soluble boron. The statistical combination of these effects results in an additional +0.01150 Δk to the reference k_{eff}, resulting in a 95/95 k_{eff} of 0.93120 with soluble boron credit. Table 6-2 contains a summary by region of the soluble boron requirements necessary to maintain $k_{eff} \leq 0.95$, as well as to account for uncertainties due to burnup and potential accidents. Column "1-1" of Table 6-2 pertains to Region 1-1.

6.1.3 Burnup Credit

Through reactivity equivalencing, the minimum burnup required, which results in the same k_{eff} as that of a fresh bundle at 1.95 $^{w}/o$, was determined. This was accomplished via depletion analysis using the CASMO-4 code. All depletion analyses incorporate conservative depletion parameters to maximize reactivity per standard practice[19]. Figure 6-1 shows the minimum burnup as a function of initial enrichment for assemblies discharged into Region 1-1. This minimum burnup curve has been increased by 4% to account for the uncertainty in calculated burnup. The minimum burnup is determined by the following equation:

$$y = ax^2 + bx + c \qquad \text{(Equation 1)}$$

where:

y = Burnup (GWD/MTU)
x = Initial enrichment ($^{w}/o$ ^{235}U)
a = -0.73864
b = 15.5634
c = -27.44461

The uncertainty in depletion dependent isotopics is conservatively assumed to be +0.01 Δk as specified in Section 5.2.1. The differential soluble boron worth was determined for each of the sub-regions, taking into account the reduced worth of boron with increasing soluble boron concentration, initial enrichment, and burnup. The reactivity effect due to depletion uncertainties was conservatively determined in the absence of soluble boron, but the limiting differential worth was used to maximize the amount of boron required. As such, the amount of boron necessary to offset a +0.01 Δk is 125 ppm.

The reactivity effect of depleting an assembly with WABAs versus an assembly without WABAs results in a Δk of +0.01 as discussed in Section 5.2.3. An additional 125 ppm is necessary under normal conditions to offset the reactivity associated with the depletion of WABAs.

The k_{eff} of a 4.5 $^{w}/o$ bundle in Region 1-1 was calculated as subsequently described in Section 6.2.1. The enrichment of the bundle was then increased incrementally to 5.0 $^{w}/o$ and the number of symmetrically distributed IFBAs required to provide the same k_{eff} (or less) for Region 1-1 was determined iteratively.

As described in Section 5.2.2, the reactivity effect due to non-uniform assembly isotopic depletion was analyzed. Non-uniform axial depletions result in an increase in Δk of +0.01345 (95/95) in Region 1-1. To mitigate this effect, an additional 167 ppm of soluble boron is required, as shown in Table 6-2.

6.1.4 Storage of Reload Fuel in Region 1-1

For fuel assemblies that do not meet the burnup versus initial enrichment requirements of Figure 6-1, including unirradiated fuel with enrichments up to 4.5 $^{w}/o$ ^{235}U (with 0 IFBA rods), storage in Region 1-1 is permitted with additional administrative controls. In this case a one out of two checkerboard configuration (with every other cell vacant) results in $k_{eff} < 1.0$ with no soluble boron and $k_{eff} \leq 0.95$ with soluble boron. Should fuel of a higher enrichment be stored in a checkerboard configuration, assemblies shall possess a minimum number of IFBA rods, as specified in Figure 6-2. To account for uncertainties in IFBA ^{10}B loading, as well as the number of IFBA rods, a conservative reactivity increase due to these effects was calculated at 0 ppm boron and the soluble boron required

determined based on the most limiting differential boron worth. To offset the reactivity effect of uncertainty in IFBA ^{10}B loading, 43 ppm of soluble boron is required. To offset the reactivity increase associated with a 10% reduction in the number of IFBA rods, 91 ppm of soluble boron is required. To mitigate an accident whereby a fresh 4.5 $^{w}/o$ ^{235}U assembly is misloaded into the checkerboard scheme, an additional 679 ppm of soluble boron is required.

Table 6-1: Region 1-1 Reactivity Changes Associated with Tolerances and Biases

Item	No Boron	With Boron
Reference (Calculation and Reactivity Equivalence Bias Corrected)		
k_{eff}	**0.96970**	**0.91970**
Tolerances and Uncertainties		
Cell ID	0.00025	0.00054
Cell Wall Thickness	0.00178	0.00101
Cell Pitch	0.00366	0.00311
Flux Trap+	0.00000	0.00000
Enrichment ($^{w}/o$ ^{235}U)	0.00731	0.00740
UO_2 Density	0.00278	0.00396
Asymmetric Position*	0.00000	0.00000
Dishing*	0.00000	0.00000
Methodology	0.00420	0.00420
Calculation	0.00040	0.00040
Total (Statistical Combination)	**0.01167**	**0.01150**
Biases		
BADGER Measurement Uncertainty+	**0.00000**	**0.00000**
Self-Shielding+	**0.00000**	**0.00000**
Upper Statistical Tolerance Limit (95/95)		
k_{eff}	**0.98137**	**0.93120**

*Negative values which are conservatively assumed to be zero.

+Not applicable because this region takes no credit for Boraflex.

Table 6-2: Soluble Boron Credit Requirements

Boron Credit	**Region**			
	1-1	**1-2**	**2-1**	**2-2**
$k_{eff} \leq 0.95$	163	371	153	231
Reactivity Equivalence				
Burnup Uncertainty (0.01 Δk)	125	125	125	125
Measured Burnup	0	0	0	0
Axial Burnup	167	167	367	244
IFBA Loading Uncertainty	43	32	0	0
IFBA Calculation Uncertainty	91	91	0	0
WABA Depletion	125	0	125	125
Accident Conditions				
Dropped Fuel Assembly or Assembly Alongside Rack	38	38	38	38
Misloaded Assembly	679	0	725	319
Abnormal Heat Load	110	110	60	60
Totals				
Total without Accident	714	786	770	725
Total with Worst Case Accident	1393	896	1495	1044

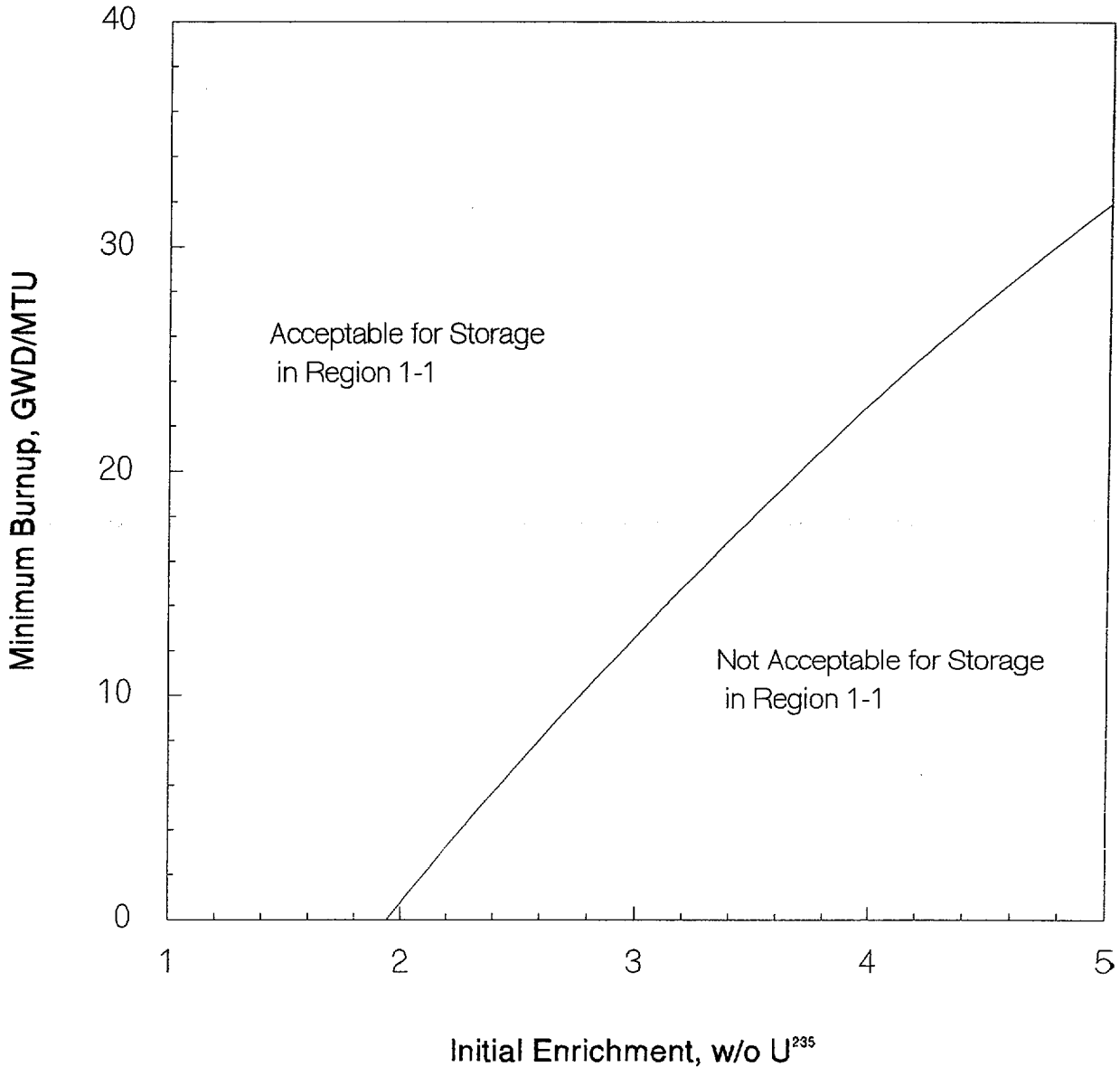

Figure 6-1: Region 1-1 Minimum Assembly Burnup - 100 % Panel Loss.

6.2 Region 1-2, 50% Boraflex Credit

6.2.1 Reference Model, Including Tolerances and Uncertainties

The maximum initial enrichment that can be safely stored in Region 1-2 is 4.5 $^{w}/o$ ^{235}U without credit for IFBA rods. A KENO V.a model of an infinite array of fuel assemblies in the X-Y direction with water reflectors above and below the fuel region results in a k_{eff} of 0.93390. This reference model assumes the rack is at a nominal pool temperature of 68°F with water at full density and no soluble boron. Table 6-3 lists the associated reactivity effects of each of the manufacturing tolerances. The statistical combination of uncertainties and tolerances adds an additional 0.00840 Δk to the reference k_{eff}, resulting in a 95% probability at 95% confidence (95/95) upper statistical limit on k_{eff} of 0.95006. This is below the limit of 1.0 without credit for soluble boron.

6.2.2 Soluble Boron Credit

In order to assure that k_{eff} remains below 0.95 with soluble boron, limited soluble boron credit is used. This lowers the 95/95 k_{eff} with soluble boron to $k_{eff} \leq 0.95$. Iterative calculations indicate that to reduce k_{eff} by 0.05 Δk, a minimum concentration of 371 ppm soluble boron is required. Column "1-2" of Table 6-2 summarizes for Region 1-2 the soluble boron necessary to maintain $k_{eff} \leq 0.95$, as well as to account for uncertainties due to burnup and accidents.

6.2.3 IFBA Credit

To accommodate the storage of higher initial enrichments up to 5.0 $^{w}/o$ ^{235}U, credit for Integral Fuel Burnable Absorbers (IFBAs) is taken. The IFBAs consists of zirconium diboride coated fuel pellets in fuel rods distributed uniformly throughout the fuel assembly. Each IFBA rod was assumed to be at the minimum loading of 1.77 mg ^{10}B / inch[21].

The IFBA equivalency plot is shown in Figure 6-2. It is noted that the minimum number of IFBAs shown in Figure 6-2 is considerably fewer than the minimum that would be used at IP2[21]. Equation 2, below, is used to determine the number of IFBA rods for fresh fuel, given the bundle initial average enrichment. Assemblies below 4.5 $^{w}/o$ ^{235}U do not require any credit for IFBA rods.

$$y=mx+b \qquad \text{(Equation 2)}$$

Where:

y = number of IFBA rods
x = Initial enrichment ($^{w}/o$ ^{235}U; range: 4.5 to 5.0w/o)
m= 48
b= -216

Table 6-3: Region 1-2 Reactivity Changes Associated with Tolerances and Biases

Item	No Boron	With Boron
Reference (Calculation and Reactivity Equivalence Bias Corrected)		
k_{eff}	**0.93390**	**0.88390**
Tolerances and Uncertainties		
Cell ID	0.00253	0.00243
Cell Wall Thickness	0.00025	0.00040
Cell Pitch	0.00163	0.00158
Flux Trap	0.00149	0.00144
Enrichment ($^{w}/o$ ^{235}U)	0.00196	0.00228
UO_2 Density	0.00210	0.00303
Asymmetric Position*	0.00000	0.00000
Dishing*	0.00000	0.00000
Methodology	0.00420	0.00420
Calculation	0.00040	0.00040
Total (Statistical Combination)	**0.00840**	**0.00859**
Biases		
BADGER Measurement Uncertainty	**0.00455**	**0.00455**
Self-Shielding	**0.00321**	**0.00321**
Upper Statistical Tolerance Limit (95/95)		
k_{eff}	**0.95006**	**0.90025**

*Negative values which are conservatively assumed to be zero.

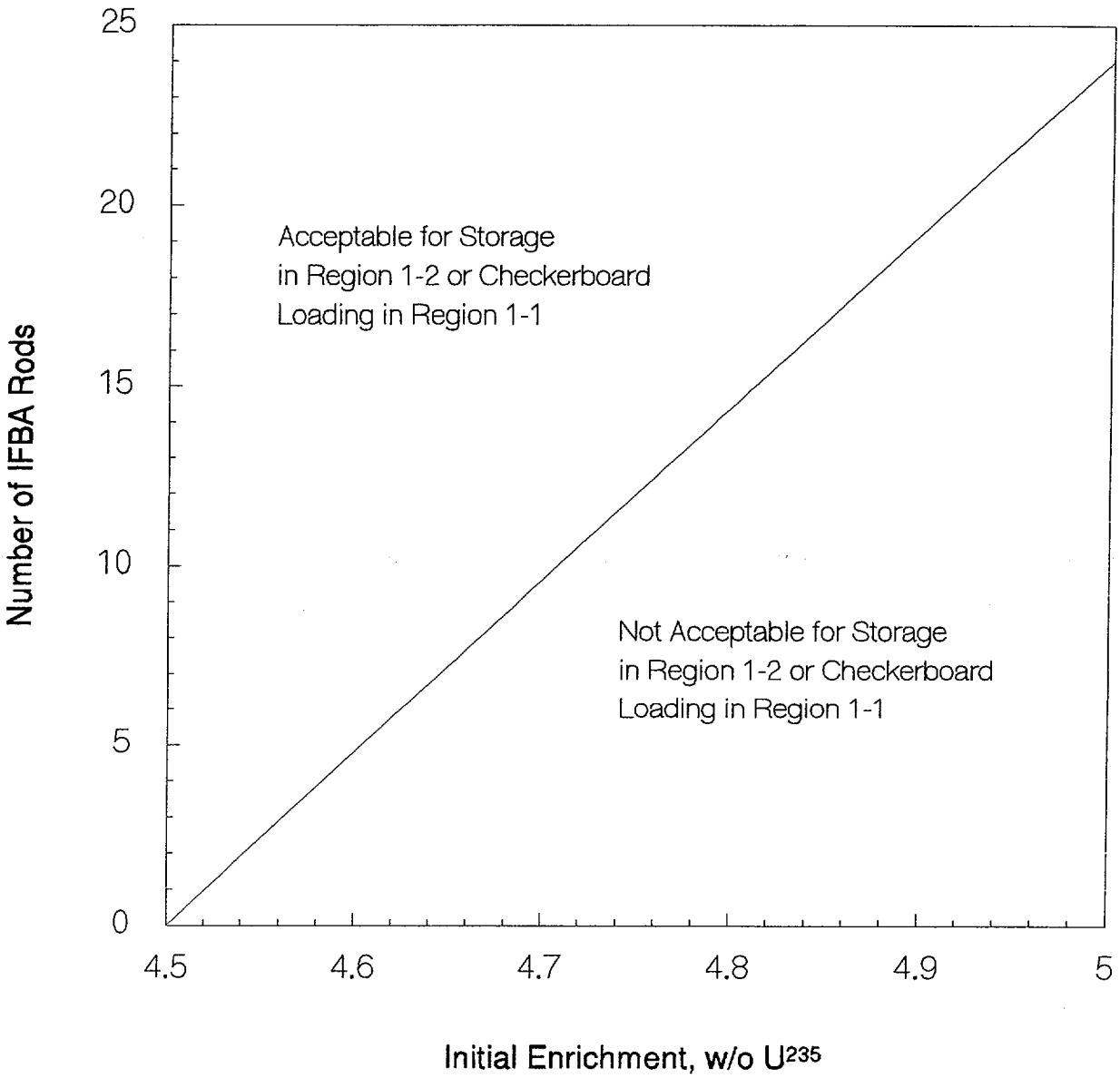

Figure 6-2: Required Minimum Number of IFBAs in Fresh Fuel versus Initial Enrichment

6.3 Region 2-1, 0% Boraflex Credit

6.3.1 Reference Model, Including Tolerances and Uncertainties

With 100% panel loss, the maximum initial enrichment that can be safely stored in Region 2-1 is 1.06 $^{w}/o$ ^{235}U with no burnup. A KENO V.a model of an infinite array of fuel assemblies in the X-Y direction with water reflectors above and below the fuel region results in a k_{eff} of 0.96910. This reference model assumes the rack is at a nominal pool temperature of 68°F with water at full density and no soluble boron. Table 6-4 lists the associated reactivity effects of each of the manufacturing tolerances. The statistical combination of uncertainties and tolerances adds an additional 0.02482 Δk to the reference k_{eff}, resulting in a 95% probability at 95% confidence (95/95) upper statistical limit on k_{eff} of 0.99392. This is below the limit of 1.0 without credit for soluble boron.

6.3.2 Soluble Boron Credit

In order to assure that k_{eff} remains below 0.95 with soluble boron, limited soluble boron credit is used. This lowers the 95/95 k_{eff} with soluble boron to $k_{eff} \leq 0.95$. Iterative calculations indicate that to reduce k_{eff} by 0.05 Δk, a minimum concentration of 153 ppm soluble boron is required. Column "2-1" of Table 6-2 summarizes for Region 2-1 the soluble boron necessary to maintain $k_{eff} \leq 0.95$, as well as to account for uncertainties due to burnup and accidents.

6.3.3 Burnup Credit

Through reactivity equivalencing, the minimum burnup required, which results in the same k_{eff} as that of a fresh bundle at 1.06 $^{w}/o$ was determined via depletion analysis with the CASMO-4 code. Since fuel in Region 2-1 may reside there for long periods of time, reactivity credit for decay of ^{241}Pu is taken into account. Figure 6-3 shows the minimum burnup as a function of initial enrichment for assemblies discharged into Region 2-1 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 6-5 contains the equations generated from best fit of the minimum burnup curves in Figure 6-3.

As described in Section 5.2.2, 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 +0.02945. This is equal to a 367 ppm soluble boron requirement, as shown in Column "2-1" of Table 6-2.

Fuel assemblies irradiated with removable WABAs can be more reactive than assemblies irradiated without WABAs after the WABAs are removed. This is attributable to spectral effects. Analysis with CASMO-4 indicates a reactivity effect of $\Delta k = 0.00951$ in the rack geometry. Conservatively attributing a bounding +0.01 Δk reactivity effect translates into a soluble boron requirement of 125 ppm as shown in column "2-1" of Table 6-2.

Table 6-4: Region 2-1 Reactivity Changes Associated with Tolerances and Biases

Item	No Boron	With Boron
Reference (Calculation and Reactivity Equivalence Bias Corrected)		
k_{eff}	**0.96910**	**0.91910**
Tolerances and Uncertainties		
Cell ID	0.00252	0.00086
Cell Wall Thickness	0.00530	0.00158
Cell Pitch	0.01396	0.01156
Enrichment ($^{w}/o$ ^{235}U)	0.01810	0.01843
UO_2 Density	0.00291	0.00443
Asymmetric Position*	0.00000	0.00000
Dishing*	0.00000	0.00000
Methodology	0.00420	0.00420
Calculation	0.00040	0.00040
Total (Statistical Combination)	**0.02482**	**0.02338**
Biases		
BADGER Measurement Uncertainty+	**0.00000**	**0.00000**
Self-Shielding+	**0.00000**	**0.00000**
Upper Statistical Tolerance Limit (95/95)		
k_{eff}	**0.99392**	**0.94248**

*Negative values which are conservatively assumed to be zero.

+Not applicable because this region takes no credit for Boraflex.

Table 6-5: Region 2-1 Minimum Burnup versus Enrichment Curves for Various Cooling Times

Cooling Time (Years)	Equation
0	$-3.0727 + 39.5461 \cdot ln(\epsilon)$
5	$-2.8395 + 34.3810 \cdot ln(\epsilon)$
10	$-2.7726 + 31.9363 \cdot ln(\epsilon)$
15	$-2.7374 + 30.5516 \cdot ln(\epsilon)$
20	$-2.7140 + 29.6705 \cdot ln(\epsilon)$

Note: the initial enrichment is designated by ϵ (as $^{w}/o\ ^{235}U$) in the table above; *ln* is the natural log function.

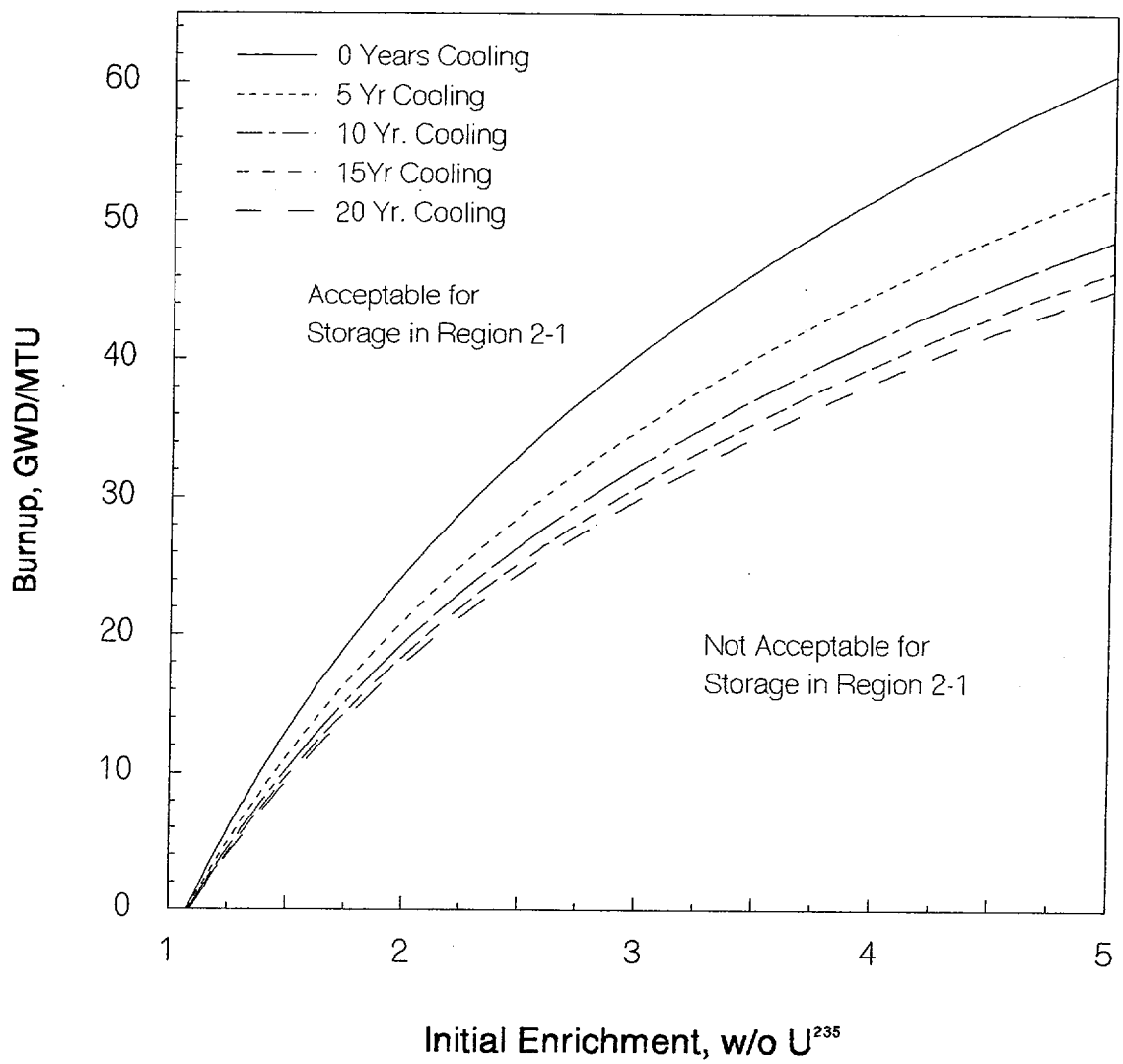

Figure 6-3: Region 2-1 Minimum Burnup Requirements versus Initial Enrichment

6.4 Region 2-2, 70% Boraflex Credit

6.4.1 Reference Model, Including Tolerances and Uncertainties

With 30% panel loss, the maximum initial enrichment that can be safely stored in Region 2-2 is 1.80 $^{w}/o$ ^{235}U with no burnup. A KENO V.a model of an infinite array of fuel assemblies in the X-Y direction with water reflectors above and below the fuel region results in a k_{eff} of 0.97365. This reference model assumes the rack is at a nominal pool temperature of 68°F with water at full density and no soluble boron. Table 6-6 lists the associated reactivity effects of each of the manufacturing tolerances. The statistical combination of uncertainties and tolerances adds an additional 0.01419 Δk to the reference k_{eff}, resulting in a 95% probability at 95% confidence (95/95) upper statistical limit on k_{eff} of 0.99598. This is below the limit of 1.0 without credit for soluble boron.

6.4.2 Soluble Boron Credit

In order to assure that k_{eff} remains below 0.95 with soluble boron, limited soluble boron credit is used. This lowers the 95/95 k_{eff} with soluble boron to $k_{eff} \leq 0.95$. Iterative calculations indicate that to reduce k_{eff} by 0.05, Δk a minimum concentration of 231 ppm soluble boron is required. Column "2-2" of Table 6-2 summarizes for Region 2-2 the soluble boron necessary to maintain $k_{eff} \leq 0.95$, as well as to account for uncertainties due to burnup and accidents.

6.4.3 Burnup Credit

Through reactivity equivalencing, the minimum burnup required, which 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 6-4 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 6-7 contains the equations generated from best fit of the minimum burnup curves in Figure 6-4.

As described in Section 5.2.2, 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 +0.01962. This is equivalent to a 244 ppm soluble boron requirement, as shown in Column "2-2" of Table 6-2.

6.4.4 Peripheral Cells

Occasionally, an assembly may be permanently discharged after only one cycle of burnup; e.g., if it has sustained mechanical failure. Select cells along the periphery of Region 2-2 (as shown in Figure 5-1) have been designated to accommodate such assemblies. These

locations along the pool wall are exempt from the minimum burnup requirements for Region 2-2. That is, fuel up to 4.5 $^{w}/o$ with zero burnup and fuel up to 5.0 $^{w}/o$ with a minimum number of IFBA rods (as shown in Figure 6-2) may be placed in these locations, as long as 1.80 $^{w}/o$ fuel (or reactivity equivalent per Figure 6-4) is in the adjacent cells. As shown in Table 6-2, 319 ppm of soluble boron is required to mitigate an accident where a 4.5 $^{w}/o$ fuel assembly is placed in the most reactive location next to the peripheral cells.

Table 6-6: Region 2-2 Reactivity Changes Associated with Tolerances and Biases

Item	No Boron	With Boron
Reference (Calculation and Reactivity Equivalence Bias Corrected)		
k_{eff}	**0.97365**	**0.92365**
Tolerances and Uncertainties		
Cell ID	0.00252	0.00099
Cell Wall Thickness	0.00100	0.00010
Cell Pitch	0.00370	0.00196
Enrichment ($^{w}/o$ ^{235}U)	0.00975	0.01005
UO_2 Density	0.00304	0.00387
Asymmetric Position*	0.00000	0.00000
Dishing*	0.00000	0.00000
Methodology	0.00420	0.00420
Calculation	0.00016	0.00016
Total (Statistical Combination)	**0.01419**	**0.01336**
Biases		
BADGER Measurement Uncertainty	**0.00502**	**0.00502**
Self-Shielding	**0.00312**	**0.00312**
Upper Statistical Tolerance Limit (95/95)		
k_{eff}	**0.99598**	**0.94515**

*Negative values which are conservatively assumed to be zero.

Table 6-7: Region 2-2 Minimum Burnup versus Enrichment Curves for Various Cooling Times

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

Note: the initial enrichment is designated by ϵ (as $^{w}/o\ ^{235}U$) in the table above; *ln* is the natural log function.

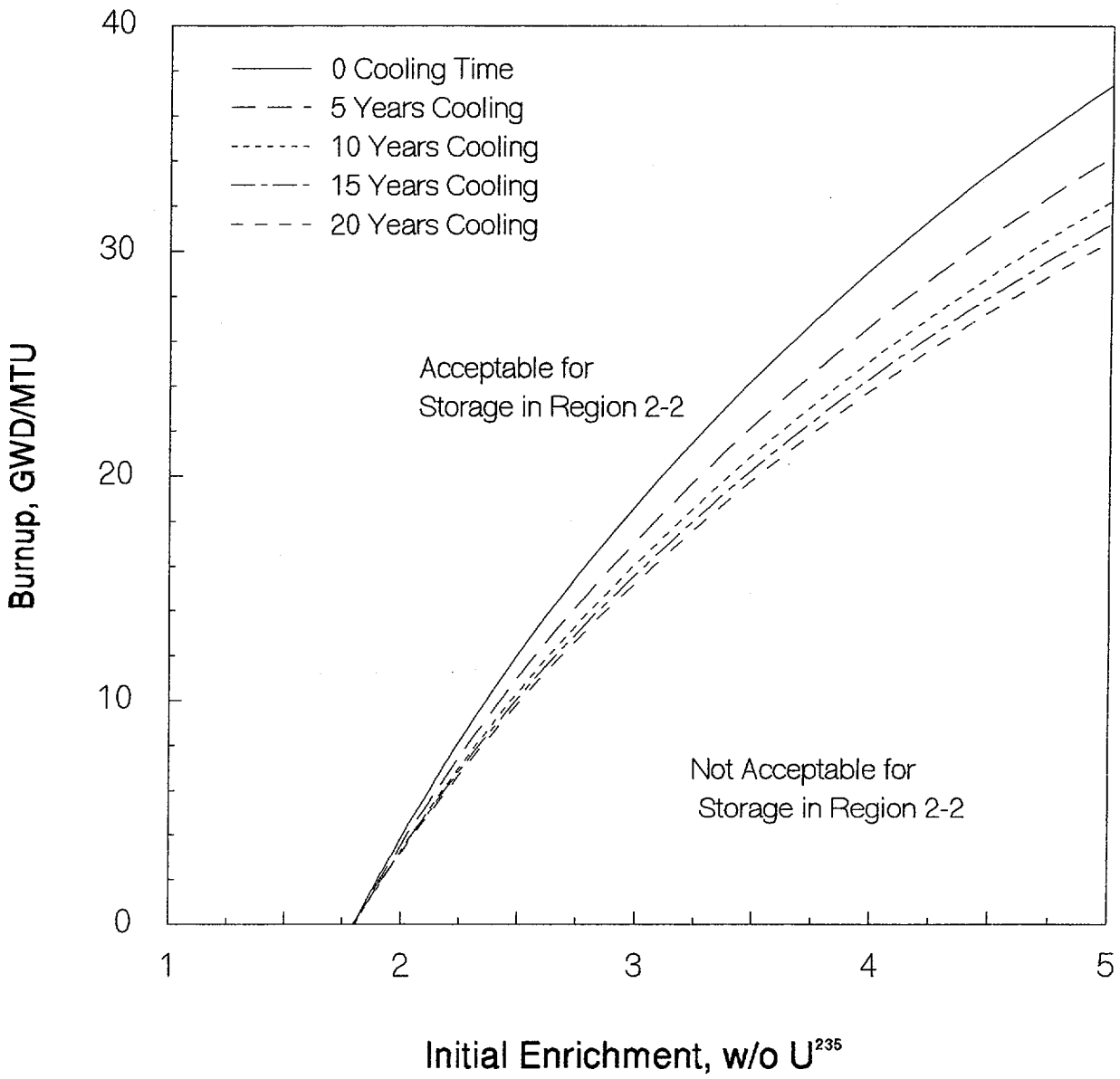

Figure 6-4: Region 2-2 Minimum Burnup Requirements versus Initial Enrichment

7.0 Accident Analysis

In addition to normal operating conditions, the occurrence of credible abnormal occurrences have been analyzed as per ANSI/ANS-57.2-1983[22], Part 6.4.2.1.B. This analysis considered the following three categories of abnormal occurrences:

- Dropped Fuel Assembly or Assembly Alongside Rack
- Misloaded Assembly
- Abnormal Heat Load

Soluble boron credit is taken for the accident condition that results in the worst condition in terms of increased reactivity. The minimum amount of additional soluble boron required is listed for each condition in Table 6-2.

The worst case for a dropped fuel assembly occurs if a fresh fuel bundle (4.5 $^{w}/o$ without IFBAs) were to be dropped on top of the Region 2 rack module G-2 (the right module in the bottom row of two modules in Figure 5-1). It was conservatively assumed that all cells contained fresh fuel at 4.5 $^{w}/o$ without IFBAs. This results in a $\Delta k = +0.00302$ increase in reactivity which is equivalent to 38 ppm soluble boron. This case bounds all dropped assembly cases for all of the regions.

The worst case in terms of a bundle alongside the racks occurs when a fresh fuel bundle (4.5 $^{w}/o$ without IFBAs) is placed in the corner of the cask area (in the lower right corner of Figure 5-1) with the Region 2-2 racks on two sides filled with fresh fuel at 1.80 $^{w}/o$ (or with equivalent reactivity). To mitigate this reactivity increase, 22 ppm of soluble boron would be required.

The worst case for a misloaded assembly occurs when a fresh bundle of maximum enrichment (4.5 $^{w}/o$ without IFBAs) were to be placed in Region 2-1 at the interface between Region 2-1 and Region 1-2. To mitigate this reactivity increase, 725 ppm of soluble boron would be required.

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. The worst-case reactivity effect due to temperature increases was in Region 1-1 and requires 110 ppm of soluble boron to mitigate the effect.

8.0 Failed Fuel Canisters

Two failed fuel canisters are loaded along Module H in the southeast corner of the pool. Module H is in the upper right corner of the pool in Figure 5-1. The failed fuel canisters are on the left side of the rectangular cutout in the upper right corner of Module H. The reactivity effect of fuel assemblies placed in the failed fuel canisters was analyzed. The Boraflex in this area of Module H is effectively intact[6,7], so that the previous analysis[1] still applies. Regardless, a worst-case situation, where 30% loss of Boraflex is assumed for Module H and fresh fuel assemblies at 4.5 $^{w}/o$ without IFBAs are placed in each canister simultaneously was considered. This results in a 95/95 increase in k_{eff} of +0.00226. This can be mitigated with an additional 28 ppm soluble boron.

9.0 Conclusions

The analyses described in this report demonstrate that under normal operating conditions, the effects of Boraflex degradation are mitigated by taking credit for fuel assembly burnup, the decay of Plutonium-241, the partial credit for soluble boron in the spent fuel pool water, and IFBA rods in the reload fuel. Without soluble boron, the effective neutron multiplication factor, k_{eff}, is less than 1.0 with a 95% probability at a 95% confidence level under normal operating conditions even when including the effects of biases, tolerances, and uncertainties. The value of k_{eff} is less than 0.95 when credit for soluble boron is taken. In the event of the worst-case accident, a soluble boron concentration of 1495 ppm would be required to mitigate the effect. IP2 Plant Technical Specifications require a minimum of 2000 ppm of soluble boron, well in excess of 1495 ppm.

Fuel assemblies with an enrichment of up to 5.0 $^{w}/o$ without burnup can be stored in Region 1-2, where 50% of the Boraflex is conservatively assumed to remain, when credit for integral fuel burnable absorbers is taken. Region 1-1 does not take credit for any Boraflex and is limited to fuel assemblies that satisfy minimum burnup criteria specified in this report. Combined with Region 1-2, these spent fuel storage cells will more than accommodate a full core offload at any time during reactor operations through 2006, as long as current assumptions about cell utilization, reactor operations, and core reload designs hold[6].

Region 2-1 does not take credit for any Boraflex and is limited to fuel assemblies that satisfy minimum burnup criteria specified in this report. Region 2-2, where 70% of the Boraflex is conservatively assumed to remain, is limited to fuel assemblies that satisfy a less restrictive set of minimum burnup criteria specified in this report. At this time, all but two discharged assemblies currently in the IP2 spent fuel storage racks can be stored in Region 2-2. These two prematurely discharged assemblies can be accommodated in any of six "peripheral" cells designated along the west side of Region 2-2. Combined with additional cells available in Regions 1-1 and 1-2 that are not required for full core offloads, these cells will accommodate all discharged fuel assemblies through 2006, as long as current assumptions about cell utilization, reactor operations, and core reload designs hold[6].

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Appendix A

BENCHMARKING OF THE SCALE-PC VERSION 4.3 CRITICALITY AND SAFETY ANALYSIS SEQUENCE SING THE KENO V.a MONTE CARLO CODE AND OF THE MULTIGROUP TWO-DIMENSIONAL TRANSPORT THEORY CASMO-4 CODE

Report No. 901-02-04

BENCHMARKING OF THE SCALE-PC VERSION 4.3 CRITICALITY AND SAFETY ANALYSIS SEQUENCE USING THE KENO V.a MONTE CARLO CODE AND OF THE MULTIGROUP TWO-DIMENSIONAL TRANSPORT THEORY CASMO-4 CODE

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Review/Approval Record

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Abstract

This report documents the benchmarking of two computer codes used in criticality analysis, SCALE-PC and CASMO-4. The first is validated with respect to a number of critical experiments to determine its bias. The second is validated with respect to a set of criticals by using bias-corrected results from SCALE-PC as benchmark values.

SCALE-PC is an Intel 486/Pentium PC modular code system for performing criticality safety analyses for licensing evaluations. The analysis sequence used in the validation includes the following modules: BONAMI-S, to prepare a problem-specific master cross section library and which performs resonance self-shielding corrections for nuclides with Bondarenko data; NITAWL-II, to prepare a working cross section library with corrections for resonance self-shielding using the Nordheim Integral Treatment; and KENO V.a, the primary module that uses the Monte Carlo method to solve the three-dimensional Boltzmann transport equation for neutron-multiplying systems. Three different cross section libraries were analyzed: 1) the SCALE 27-energy group ENDF/B-IV library (27GROUPNDF4), 2) the 218-energy group ENDF/B-IV library (218GROUPNDF4) and 3) the 238-energy group ENDF/B-V library (238GROUPNDF5). CASMO-4 is a multigroup two-dimensional transport theory code which calculates a deterministic value for k_∞. It has its own standard cross-section library.

Five fuel rod array critical experiments performed by Babcock and Wilcox (B&W) and eight critical experiments selected by the International Committee on the Safety of Nuclear Installations (CSNI) were modeled and executed to determine the bias in the SCALE-PC code calculation of k_{eff}. These represent a variety of fuel enrichments, neutron absorber strengths, and $H/^{235}U$ ratios.

The central fuel array of the five Babcock and Wilcox experiments were also modeled using CASMO-4. These were compared with SCALE-PC models using the exact geometry modeled for CASMO-4.

The results show that SCALE-PC and CASMO-4, when implemented as directed in this document to include a correction for a quantifiable negative bias, provide a k_{eff} or k_{∞} that is suitable for application to calculating the reactivity state of typical LWR fuel/rack and fuel/cask configurations.

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1.0 INTRODUCTION

The purpose of this report is to document the benchmarking of two computer codes used in the criticality analysis of nuclear fuel assemblies in storage or shipping cask arrays (i.e., to determine k-effective or k-infinity of systems containing fissile material). These two codes are SCALE-PC Version 4.3[1], an Intel 486/Pentium PC code used at NETCO and containing the Monte Carlo code KENO V.a, and CASMO-4[2], which can be executed on mainframe computers or workstations.

SCALE-PC is a collection of computer codes and cross section libraries used to perform criticality safety analyses for licensing evaluations. In particular SCALE-PC uses the Monte Carlo code KENO V.a to solve 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; and NITAWL-II to prepare working cross section libraries with corrections for resonance self-shielding using the Nordheim Integral treatment. CASMO-4 is a two-dimensional multi-group transport theory code for fuel assembly burnup analysis in-core or in typical fuel storage rack or cask geometries.

To validate and verify these codes for use in spent fuel rack and shipping cask design evaluations, the SCALE-PC codes were used to model a number of critical experiments. The calculated k_{eff} was compared to the critical condition ($k_{eff} = 1.0$) to determine the bias in the calculated values. In the SCALE-PC calculations three cross-section libraries were analyzed: 1) the 27 energy group ENDF/B-IV cross-section library, 2) the 218 group ENDF/B-IV library and 3) the 238 energy group ENDF/B-V cross-section library. Benchmarking of the 27 group library was documented previously[3] while the latter two libraries were analyzed in an attempt to reduce the methodology bias. Subsequently, both SCALE-PC and CASMO-4 (with its own 70 energy group cross section library) were used to model central arrays of critical experiments. It is noted that CASMO-4 is a cell code in which infinitely repeating arrays of fuel assemblies and/or fuel racks are modeled. 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.

A set of five Babcock and Wilcox (B&W) critical experiments, XIII, XIV, XV, XVII, and XIX[4], were selected because they closely represent typical fuel/rack geometries with neutron absorber panels. The International Committee on the Safety of Nuclear Installations (CSNI) identified a sequence of benchmark problems[5] that cover both fuel/rack and fuel/cask geometries, and were representative of various enrichments and $H/^{235}U$ ratios. The resulting models are representative of most fuel storage rack and fuel cask configurations used today. The critical experiment input files were taken from Reference [3] which verified and validated an earlier version of SCALE-PC

Section 2 of this report describes the initial verification of SCALE-PC conducted by executing a set of standard problems supplied with the binary executable code. The code is distributed by the Radiation Shielding Information Center (RSIC), and is designated as Code CCC-545. The CASMO-4 problems were executed on a RISC workstation by Studsvik of America under contract to NETCO. The input files were developed by NETCO and forwarded to Studsvik for execution. The output files were returned to NETCO for interpretation and evaluation. All work was performed under NETCO's Quality Assurance Program[6].

In Section 3.1 the results of the SCALE-PC modeling of the B&W and CSNI critical experiments to calculate the SCALE-PC bias are covered. In Section 3.2 SCALE-PC results are compared with CASMO-4 transport theory calculations for the five B&W critical experiments modeled as infinite arrays. This allows the CASMO-4 bias to be computed based on the SCALE-PC bias determined in Section 3.1. In addition, these comparisons

independently verify each of the codes with respect to the other.

The methods benchmarking described in this report have been patterned after those in industry accepted standards[7,8].

2.0 BENCHMARKING - STANDARD PROBLEMS AND CONFIGURATION CONTROL

2.1 SCALE-PC Configuration Control

The binary executable codes and associated batch files were provided by RSIC on CD-ROM for use on Intel 486/Pentium based micro-computers running under the DOS 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 a SCALE-PC sequence the user will verify the file names, creation dates, and sizes to insure that they have not been changed. Appendix B contains the CD-ROM which include the as-received version of all files required to execute this program. 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 SCALE-PC Sample Problems

A series of input files with their corresponding output files were provided with the 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 RSIC and the resulting output files compared to those provided by RSIC on CD-ROM. Except for the date and time of execution stamps, the respective output files were identical. The KENO V.a module of SCALE-PC 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 version of SCALE-PC is identical to the version documented in the User's Manual[1].

Examination of the sample input decks shows that the run modules in batch file "CSASKENO.BAT" exercise all of the code options used by this benchmarking exercise. Before and after each subsequent use of SCALE-PC this one set of sample input modules will be executed and the output files compared to the sample output files (CSAS.OUT and KENOVA.OUT) 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

As noted previously, the version of CASMO-4 used for this analyses was developed for a RISC workstation. Version 2.05.01 of CASMO-4 was used for this benchmarking work and subsequent users of CASMO-4 for NETCO will verify that Version 2.05.01 is being used. CASMO-4 and all versions are controlled by Studsvik of America under their Quality Assurance Program[9]. If a different version of CASMO-4 is used by NETCO for any subsequent analyses, the CASMO-4 analyses in Section 3.2 shall be repeated with the version in use.

3.0 BENCHMARK MODELING OF LWR CRITICAL EXPERIMENTS

An index of input and output files for each experiment modeled is contained in Appendix C. For each experiment, the input and output files are on 3.5 inch 1.44 MB diskettes which are also contained in Appendix C. Appendix D 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-PC

The B&W experiments[4] include twenty water moderated LWR fuel rod cores and close-packed critical LWR fuel storage arrays. Of these, five 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 experiments have been modeled as they most closely represent LWR fuel in high density fuel storage rack and cask configurations with neutron absorber panels. Table 3-1 summarizes some of the model parameters including U-235 enrichment, moderator-to-fuel-ratio and absorber macroscopic absorption cross-section.

The Committee on the Safety of Nuclear Installations (CSNI) have published a selection of critical experiments[5] 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 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 reflected arrays but are at a higher enrichment (4.75 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 4.75 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 numerical model of the B&W criticals 2,000,000 neutrons in 2,000 generations were tracked after skipping the first 30 generations to insure source convergence. In each numerical model of the CSNI criticals at least 1,000,000 neutrons in at least 1,000 generations were tracked after skipping at least 20 generations to insure source convergence. The output files were always checked to insure 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 given 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 SCALE-PC bias is calculated as follows. First, the variance-weighted mean is calculated as (3-1)

$$k_m = \sum_{i=1}^{N} (k_i/\sigma_i^2) \Big/ \sum_{i=1}^{N} (1/\sigma_i^2)$$

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

critical i. The standard deviation around k_m is given by

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

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. The resulting mean bias for this library is -0.0087±0.0042

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 3.1% whereas for the 27 Group ENDF/B-IV library it was 4.4% and for the 218 ENDF/B-IV library it was 0.01% 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 15.5% while for the 27 Group ENDF/V-IV library it was 34.8% and for the 218 Group ENDF/B-IV library it was 54.7%. 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 [4] and CSNI [5] Critical Experiments - Design Parameters

Reference	Experiment Number	Absorber Type	Absorber Σ_a [cm^{-1}]	Enrichment w%	$H/^{235}U$ Ratio
4	XIII	BORAL	1.871	2.459	216.43
4	XIV	BORAL	1.460	2.459	216.52
4	XV	BORAL	0.475	2.459	216.52
4	XVII	BORAL	0.293	2.459	216.54
4	XIX	BORAL	0.129	2.459	216.54
5	1-1	none	-	2.35	398.72
5	1-2-1	none	-	4.75	109.44
5	1-2-2	none	-	4.75	228.53
5	2-1	BORAL	30.6	2.35	398.72
5	2-2	BORAL	24.6	4.75	228.53
5	3-A-1	none	-	2.35	398.75
5	3-B-1	none	-	2.35	398.75
5	3-A-2	BORAL	24.6	4.75	228.53

Table 3-2 B&W[4] and CSNI[5] Critical Experiment Results

Reference	Experiment	27GROUPNDF4			218GROUPNDF4			238GROUPNDF5		
		k_{eff}	sigma	bias	k_{eff}	sigma	bias	k_{eff}	sigma	bias
4	XIII	0.99080	0.00046	-0.00920	0.98722	0.00050	-0.01278	0.99406	0.00050	-0.00594
4	XIV	0.98768	0.00046	-0.01232	0.98350	0.00048	-0.01650	0.99079	0.00049	-0.00921
4	XV	0.98415	0.00043	-0.01585	0.98004	0.00046	-0.98622	0.98622	0.00045	-0.01378
4	XVII	0.98640	0.00043	-0.01360	0.98278	0.00045	-0.01722	0.99025	0.00045	-0.00975
4	XIX	0.98866	0.00042	-0.01134	0.98347	0.00043	-0.01653	0.99070	0.00043	-0.00930
5	1-1	0.99034	0.00047	-0.00966	0.98735	0.00047	-0.01265	0.99015	0.00048	-0.00985
5	1-2-1	0.99431	0.00070	-0.00569	0.99060	0.00072	-0.00940	0.99216	0.00069	-0.00784
5	1-2-2	0.99875	0.00060	-0.00125	0.99598	0.00063	-0.00402	0.99582	0.00064	-0.00418
5	2-1	0.98750	0.00047	-0.01250	0.98683	0.00050	-0.01317	0.98851	0.00049	-0.01149
5	2-2	0.99524	0.00067	-0.00476	0.99263	0.00071	-0.00737	0.99448	0.00071	-0.00552
5	3-A-1	0.99132	0.00059	-0.00868	0.98998	0.00067	-0.01002	0.99387	0.00067	-0.00613
5	3-B-1	0.98956	0.00064	-0.01044	0.98815	0.00064	-0.01185	0.99133	0.00065	-0.00867
5	3-A-2	0.99887	0.00069	-0.00113	0.99777	0.00056	-0.00223	0.99936	0.00071	-0.00064
			bias	-0.0102		bias	-0.0130		bias	-0.0087
			Sigma	0.0047		Sigma	0.0055		Sigma	0.0042

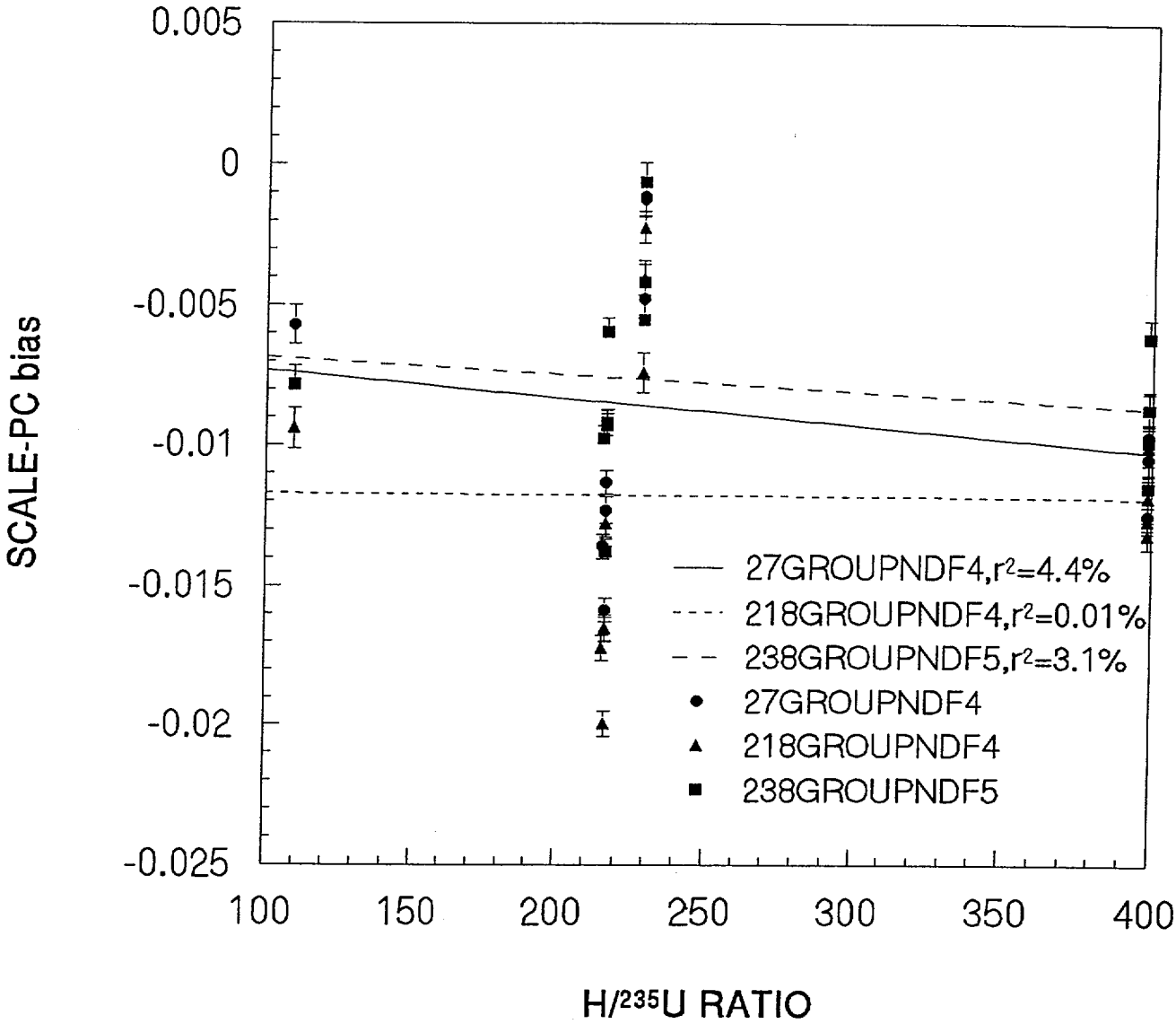

Figure 3-1: Variation of SCALE-PC bias with Moderator-to-Fuel Ratio

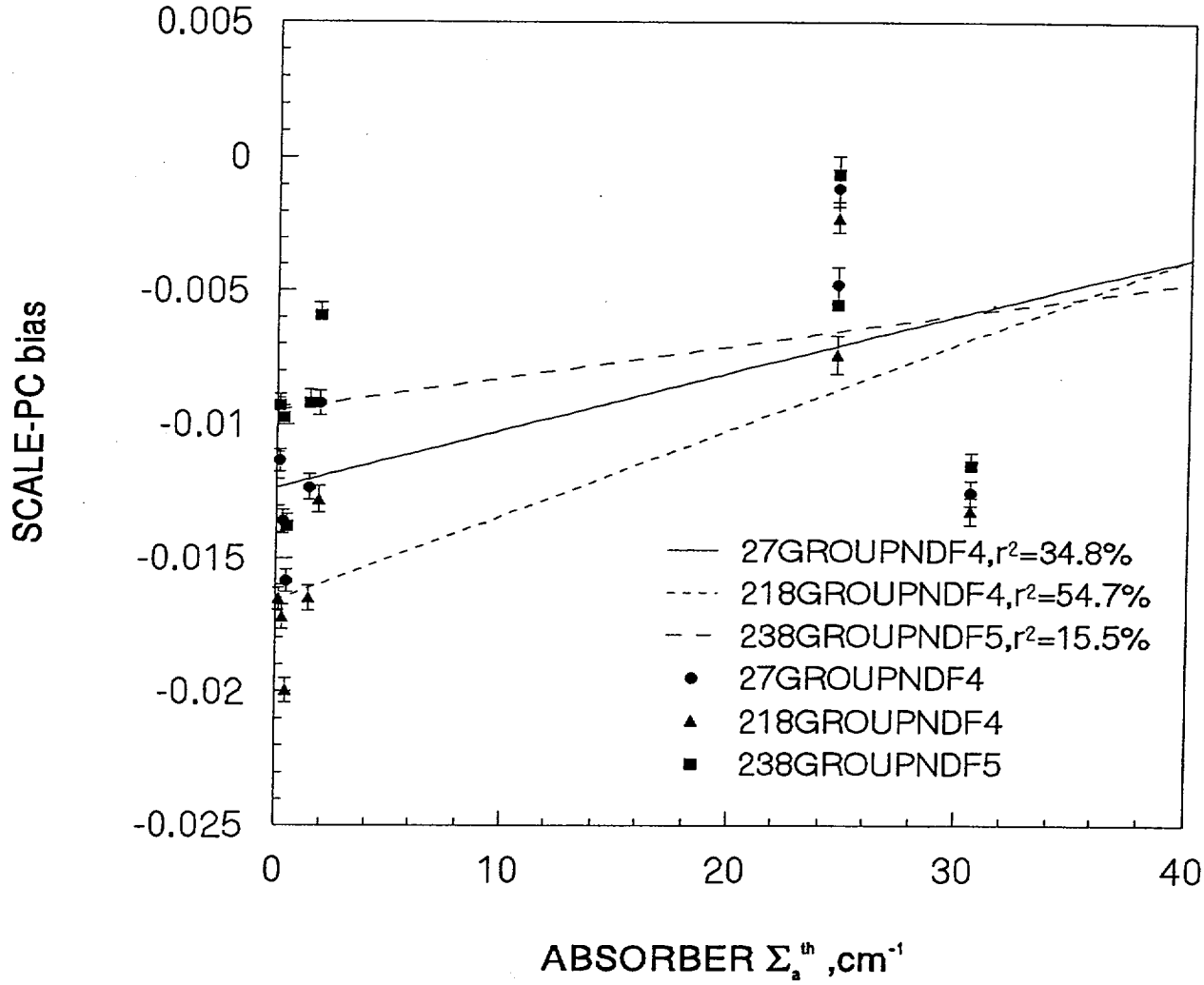

Figure 3-2: Variation of SCALE-PC Bias with Absorber Strength

3.2 BENCHMARKING OF CASMO-4

This section compares SCALE-PC[1] and CASMO-4[2] calculations for k_∞ of the same five B&W critical experiments[5] discussed in Section 3.1. CASMO-4 is limiting 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-PC 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-PC is known (see Section 3.1), it can be applied to the SCALE-PC 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-PC and CASMO-4 analyses are compared in Table 3-3. The CASMO-4 bias is calculated as

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

where

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

The resulting mean bias and standard deviation for the 238 Group ENDF/B-V library are -0.0113 ± 0.0024.

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

Experiment	CASMO-4	SCALE PC(bias corrected)								
		27GROUPNDF4			218GROUPNDF4			238GROUPNDF5		
		k_{eff}	sigma	bias	k_{eff}	sigma	bias	k_{eff}	sigma	bias
XIII	1.08947	1.10050	0.00083	-0.01103	1.09882	0.00091	-0.00935	1.10239	0.00050	-0.01292
XIV	1.08993	1.10048	0.00084	-0.01055	1.09819	0.00090	-0.00826	1.10383	0.00049	-0.01390
XV	1.09898	1.10661	0.00073	-0.00763	1.10628	0.00081	-0.00730	1.11112	0.00045	-0.01214
XVII	1.1077	1.11447	0.00073	-0.00677	1.11408	0.00079	-0.00638	1.11685	0.00045	-0.00915
XIX	1.11607	1.12127	0.00520	-0.00520	1.11955	0.00074	-0.00348	1.12446	0.00043	-0.00839
			bias	-0.0082		bias	-0.0070		bias	-0.0113
			Sigma	0.0025		Sigma	0.0020		Sigma	0.0024

4.0 CONCLUSIONS

SCALE-PC has been benchmarked by modeling five Babcock and Wilcox critical experiments and eight CSNI critical experiments representative of fuel storage rack and fuel cask geometries. It is recommended that the 238 energy group ENDFB/V library be used for all production criticality analyses. This library has the smallest negative bias of -0.0087± 0.0042.

The reduced bias of the 238 group ENDF/B-V library relative to the 27 Group and 218 Group ENDF/B-IV libraries is postulated to be a result of the improvements to NITAWL-II interpolation of thermal scattering matrices as well as the availability of additional thermal scattering data in the ENDF/B-V libraries. In previous versions of SCALE, NITAWL-II lacked the interpolation capability and would select cross-section data closest to the temperature input by the user. Negative Biases of up to 2% have been reported in the references for the 27 group and 218 group ENDF/B-IV libraries. These biases have been found to be a result of inadequate resonance cross sections for U-238, which constituted one of the major revisions between the ENDF/B-IV and ENDF/B-V cross section libraries. It was determined that there is no significant bias dependence on the moderator-to-fuel ($H/^{235}U$) ratio or absorber strength.

CASMO-4 has also been benchmarked by modeling the five Babcock and Wilcox critical experiments as infinite arrays. Best estimates of the k_∞ for the exact same geometry were calculated using SCALE-PC and applying the mean bias reported above. The CASMO-4 bias with respect to these values was calculated to be -0.0113 ±0.0024 (1 sigma). The comparison of SCALE-PC and CASMO-4 serves to verify the results of each with respect to the other.

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ATTACHMENT 4 TO NL 01-110

NET-173-02
INDIAN POINT UNIT NO. 2 SPENT FUEL POOL (SFP) BORON DILUTION ANALYSIS

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247

NET-173-02

Indian Point Unit 2 Spent Fuel Pool (SFP) Boron Dilution Analysis

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Review/Approval Record

Revision	Date	Prepared by:	Reviewed by:	Quality Assurance:
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1	9/12/01	[illegible]	M. Harris	L.P. Mariani

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1 INTRODUCTION

Previously, criticality analyses for the Indian Point 2 (IP2) spent fuel pool had depended upon crediting the Boraflex present in the spent fuel racks. Due to a premature deterioration of this material, it has become necessary to take credit for soluble boron contained in the spent fuel pool (SFP) water to maintain criticality criteria. Based upon concurrent analyses, a minimum acceptable soluble boron concentration has been determined which would mitigate a worst-case fuel loading accident. This report documents the evaluation of potential accidents that could cause significant dilution of the SFP by adding large volumes of unborated water. A worst-case dilution scenario has been identified and a minimum SFP boron concentration has been established. This will ensure that credit may be taken for soluble boron under all reasonable accident conditions.

The present analysis is deterministic in nature; it provides a quantitative analysis of all plant specific events to determine the maximum time necessary to detect and mitigate the worst dilution event. A generic probabilistic analysis of boron dilution, Ref. [1] WCAP-14181, was previously carried out by the Westinghouse Owners Group.

The dilution analysis examines the various paths/scenarios by which the SFP boron concentration may be diluted and the time available before the minimum boron concentration, necessary to ensure subcriticality under worst case accident conditions, is reached. This minimum boron concentration has been calculated to be 786 ppm boron, Ref. [4], for the non-accident condition – i.e. it is not assumed an assembly is misloaded concurrent with the spent fuel dilution accident.

An analysis was performed for various possible boron dilution events and included a walkdown of the IP2 SFP. During the process of this analysis, several events were eliminated because they were not credible or did not lead to pool dilution. Those events that were deemed credible and applicable are outlined below; their severity is quantified through a calculation of the time required to dilute the pool below the minimum acceptable boron concentration.

The present analysis assumes the IP2 Technical Specification limit (3.8.D.2) on the frequency of testing the SFP boron concentration, and required boron concentration, will be revised to once per week and 2000 ppm boron, respectively.

2 SPENT FUEL POOL (PIT) AND FUEL STORAGE BUILDING SYSTEMS RELATED TO BORON CONTROL

The following provides information relevant to the potential for boron dilution and fuel storage building system features that aid in its detection.

2.1 SPENT FUEL POOL BUILDING, POOL, RACKS, AND SPENT FUEL

The fuel storage building consists of the spent fuel pit constructed of reinforced concrete and founded on rock and the steel superstructure above the pit, which encloses the pit and supports the fuel cask handling crane.

The refueling canal and spent fuel storage pit are reinforced concrete structures with a seam-welded stainless steel plate liner. These structures are designed to withstand any anticipated earthquake loadings (for Operational Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE)) as seismic Class I structures so that the liner should prevent leakage even in the event the reinforced concrete develops cracks. The building superstructure was designed as a Class III structure.

The spent fuel assemblies and spent fuel racks need not be described here in detail. However, it is noted that the spent fuel (assuming racks are filled to capacity) and spent fuel rack's volumes have been included in a conservative calculation of the spent fuel pool water volume. It is further assumed that the racks and fuel are of a sufficiently open nature so that bulk mixing will ensure the local differences in boron concentration will be negligible (the mixing will also be enhanced due to the natural circulation associated with spent fuel assembly decay heat).

According to Ref. [5], the Spent Fuel Pool (SFP) water volume is conservatively calculated as 33,000 cu ft. and the pool surface area is 960 sq ft.

2.2 INSTRUMENTATION

Of principal significance is the SFP level instrumentation. This is a high and low spent fuel [Pit] level, which is indicated and alarmed in the Central Control Room. High-Low alarm is a float switch assembly set for ± 6" of normal level, Ref. [8].

Assuming the water level is at the low level alarm setpoint, the maximum volume of liquid that could be added, before a high level alarm occurs, is 7181 gallons. Additionally, with the water at the minimum level, an addition of 12,567 gallons would result in the pool water over flowing.

2.3 SFP REQUIRED MINIMUM BORON CONCENTRATION AND SAMPLING FREQUENCY, AND ADMINISTRATIVE PROCEDURES

The current Tech Spec (3.8.D.2) minimum Boron is 1500 ppm and sampling frequency is monthly--with maximum of 45 days.

> For the present review/analysis it is assumed the IP2 Tech Spec will be changed to sampling once every seven (7.0) days, and the minimum soluble boron concentration changed (increased) to 2000 ppm.

In addition to soluble boron concentration, the SFP water level is verified once per 12-hour shift, at approximately equal intervals (i.e., the water level is visually checked--to verify it is between the upper and lower float switch positions).

Thus, significant leakage into the pool would normally be detected, after a negligible delay, by level instrumentation and after about 12 hours by operator verification of water level.

2.4 BORATION SOURCES

The boron concentration is increased by: (1) direct addition of boric acid into the SFP and circulating water via the spent fuel pit cooling pump; (2) bleed and feed from the Refueling Water Storage Tank (RWST) and circulating water via the spent fuel pit cooling pump; and (3) connecting a temporary hose from the Chemical Volume Control System (CVCS) blender to the SFP.

2.5 SFP COOLING LOOP

The SFP cooling loop is Class II. The cooling loop consists of two pumps (one redundant), heat exchanger, filter, demineralizer, piping, and associated valves and instrumentation. One of the pumps draws water from the pit, circulates it through the heat exchanger and returns it to the pit. Component cooling water cools the heat exchanger. A portion (about 5%) of the pump discharge flow is passed through the SFP demineralizer and filter to maintain water purity and clarity before being returned to the pit (see Section 2.7).

The spent fuel pit heat exchanger is of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pit water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel. One of two spent fuel pit pumps circulates water in the spent fuel pit cooling loop. The second pump is on standby.

2.6 SFP MAKEUP WATER AND POTENTIAL DRAIN PATHS

The SFP is filled from the RWST. Technical specification 3.7.A.1.a requires that the SFP boron concentration be maintained at or above 2000 ppm. The refueling water purification pump (100 gpm) circulates RWST water through the SFP purification loop.

Makeup water is from the Primary Water System (PWS). The primary water makeup pumps take suction from the Primary Water Storage Tank (PWST) and provide non-borated water to the blender.

There are 2 primary water make-up pumps, each rated at 150 gpm at 210 feet Total Dynamic Head (TDH). These pumps are controlled from the control room; normally one pump is running and the second pump is not running but will start on receipt of a signal from the RCS auto make-up control. The switch operating these pumps is in the control room. They receive water from the 165,000 gallon primary water storage tank. The tank level is displayed in the control room, as is the (common) high and low level alarms.

2.7 SFP PURIFICATION LOOP

A portion (about 5%) of the pump discharge flow is passed through the SFP

demineralizer and filter to maintain water purity and clarity before being returned to the pit. A separate pump is used to circulate RWST water through the same demineralizer and filter for purification.

When it is required to clean up the RWST, the refueling water purification pump circulates water in a loop between the RWST and SFP demineralizer and filter. The SFP filter and strainer remove particulate matter. About 5% (about 100 gpm) of the SFP pump discharge flow can be directed to the demineralizer. This loop allows the SFP level to be adjusted in concert with the primary water system or RWST.

A separate pump is used to circulate Refueling Water Storage Tank (RWST) water through the same demineralizer and filter for purification.

3 METHOD OF CALCULATING DILUTION TIMES AND VOLUMES

The freeboard to the top of the SFP is about 4% of the total SFP liquid volume, and this additional water is conservatively neglected. Accordingly, the addition of any water to the initial SFP water volume (at the alarm setpoint), is assumed to immediately overflow the SFP. This results in boron dilution times that are about 4% conservative.

Not withstanding the above, the effect of the finite time and volume to reach the high level alarm, and the time and volume for the water to begin to overflow, are calculated. This is done to characterize the time until the SFP overflows, and the resulting boron dilution at that time. As regards calculating the time that the soluble boron concentration reaches the minimum acceptable value (786 ppm), the assumption is made that the water immediately overflows, even though its actual level will be anywhere from 1.5 to 0.5 feet below the top of the pool.

Consistent with the above assumption, the standard dilution expression is obtained from conservation of mass for a well mixed fluid volume. Assuming inflow equals outflow, expressing the equation in terms of the boron concentration, and integrating from C(0) to C(t) and t=0 to t, solving for t yields:

$$t = [V(0)/Vdot]\ln[[C(0)-CIN]/[C(t)-CIN]]. \qquad (3\text{-}1)$$

If the total diluting volume (VIN=Vdot*t) is finite, the final concentration $C(\infty)$ can be calculated. i.e.,

$$C(\infty) = CIN + [C(0)-CIN]\exp[-VIN/V1] \qquad (3\text{-}2)$$

In addition, the change in the SFP boron concentration during the initial period of water addition, prior to the pool water over flowing, is given by,

$$C(t) = [V(0)*C(0) + V1*CIN]/[V(0) + V1] \qquad (3\text{-}3)$$

The definitions of the variables are as follows:

C(t) = boron concentration in SFP [ppm]
C(0) = initial boron concentration [ppm]
C(∞) =concentration after a finite volume (e.g., small tank) is completely added to the SFP [ppm]
CIN = incoming boron concentration (normally CIN = 0) [ppm]
t =elapsed time (t > 0) [min]
Vdot =volumetric (dilution) flow into SFP, at CIN [gpm]
V(0) = initial liquid volume in SFP [gal]
V1 = volume of water added to the SFP, to reach the top of the SFP, at CIN.[gal]
VIN = total diluting volume (= Vdot*t) [gal]

3.1 SELECTED ANALYSIS RESULTS

The criticality analysis results in a soluble boron concentration of 786 ppm to ensure subcriticality under the non-accident conditions. Calculations were performed to determine the times necessary for the boron concentration to dilute to this value. For an initial SFP boron concentration of 2000 ppm, the resulting time to reach 786 ppm is given,

t [day] = 160.105 [gal-day/min]/Vdot [gpm]

or, Vdot [gpm] = 160.105 [gal-day/min]/t [day]

Some typical times (to dilute from 2000 to 786 ppm) are provided below.

T (2000->786):	7 days	2 days	1 day	12 hours	8 hours
Vdot (gpm):	22.9	80.1	160.1	320.2	480.3

The total water volume that would reduce the concentration from 2000 to 786 ppm boron is 230,551 gallons. i.e.,

Vdot[gpm]*t[day]=160.105 [gal-day/min] * 24x60 [min/day] = 230,551 gallon

In addition, the maximum amount of water that could be added before high level alarm actuation is 7181 gallons, and the maximum water addition that could occur prior to the pool water over flowing is 12,567 gallons. These readily observable events would occur prior to significant boron dilution. The boron concentration is reduced (from 2000 ppm) to 1943 ppm and 1916 ppm, respectively. Thus the readily observed indications of high water level (alarm and then water over flowing the pool) would occur prior to any significant dilution of the boron concentration.

For purposes of screening/characterizing dilution event severity, it is convenient to utilize the following calculation approach. As an example, assume a dilution (event) with a 150 gpm flow rate is identified 20 hours after water begins to overflow the SFP. For this event the time available for the operator to respond (to terminate and/or mitigate the event) is given by,

$T_{RESPOND}$ = (160.105-(12,567/60/24) [gpm-day])/Vdot - 20/24 [day]
= 0.18 day = 4.4 hour

4 DILUTION PATHS

The major dilution event not associated with a pipe break or rupture of a heat exchanger tube, is the inadvertent dilution of the SFP with the primary water make-up water pumps drawing unborated water from the primary water tank.

4.1 DILUTION FROM PRIMARY WATER MAKE-UP SYSTEM

This event corresponds to the inadvertent dilution of the SFP with the primary water make-up water pumps drawing unborated water from the primary water tank. The Primary Water Storage Tank (PWST) has a nominal water volume of 165,000 gallons, Ref. [6]. The maximum possible water volume of the PWST, up to the overflow tube is 177,479 gallons, Ref. [6] and Ref. [9]. This conservative water volume is insufficient to dilute from 2000 to 786 ppm.

5 PIPE BREAKS AND LEAKS

The following identifies the potential dilution mechanisms due to the mechanical failure of piping and tubing and quantifies the times for the boron concentration to decrease from 2000 ppm to 786 ppm.

5.1 LEAKS FROM OUTSIDE THE SFP BUILDING

The potential exists for a SFP heat exchanger tube leak--thus allowing component cooling water (CCW) to enter the SFP cooling loop; however, the make-up water for the CCW is the PWST. Thus, as discussed in section 4.1, the water volume is insufficient to dilute from 2000 to 786 ppm boron.

5.2 LEAKS FROM INSIDE THE SFP BUILDING

The top of the SFP is at elevation 95 foot, and there are several potential sources of water that can flow into the SFP--i.e., sources that are at elevation 95 foot and above. These include: (1) steam heating and condensate return piping, (2) fire protection standpipe, (3) city water, for clean up, etc. (4) primary water, and (5) precipitation (water) in-leakage from the roof.

These sources were identified/characterized from oral commentary by Con Edison personnel, review of drawings and other documents, and a walk through of the SFP building--in the vicinity of the top of the SFP.

5.2.1 Steam Heating and Condensate Return Leakage

Two (2) building heaters receive steam and return condensate. The total capacity is 260,000 Btu/hr/heater (20 psig steam) and it is conservatively assumed that a single line may carry the total flow to/from both heaters. It is assumed the pipe may suffer a complete rupture (e.g., due to water hammer) and the resulting long term leakage is 0.9 gpm.

For a leakage flow of 0.9 gpm, it would take 178 days for the pool to dilute from 2000 ppm to 786 ppm boron. This is large compared to the SFP boron sampling frequency. Thus, even without credit for the high level alarm and SFP overflowing, the leakage would be detected from the decrease in soluble boron concentration.

5.2.2 Fire Protection Standpipe Leakage

The fire protection pipe is a 4" diameter standpipe, at the 95 foot elevation (also on the SFP building roof). It is a schedule number 40 pipe, with a 4.026" inner diameter and a 0.237 inch wall thickness. The normal fire main pressure is kept at 135 psi (relief valves discharge to the city water header at 142 psig, and two others at 147 psig). For this analysis, a moderate-energy piping failure is assumed.

For moderate-energy pipes, the US NRC Branch Technical Position, Ref. [2] and Ref. [3], is utilized. Specifically, a crack area is assumed to be a circular area, given by (1/2)*(pipe diameter)*(1/2)(wall thickness). The resulting flow area has a thickness-to-width ratio of 0.43. It is very thick, and (for turbulent flow) can be considered as a sudden contraction (K = 0.5) followed by a sudden expansion (K = 1.0), or total K = 1.5 (equivalent to a discharge coefficient of 0.816).

However, conservatively a loss coefficient of unity is used (K = 1.0), resulting in a leakage flow of 108 gpm. (Had a K value of 1.5 been used, the flow is calculated to be 88 gpm.)

A leakage of 108 gpm would result in 35.6 hours to dilute the SFP from 2000 to 786 ppm boron. Although this leakage would not be detected by the required boron concentration testing frequency, the leakage is large and would be easily observed during the required operator rounds which occur every 12 hours, and during normal security rounds which typically occur every couple of hours. The pool water would begin to overflow 116 minutes after the inception of the piping failure.

The operators verify the SFP level at 8:00 AM and 8:00 PM, and conservatively assuming the operator inspections are 4 hours early and 4 hours late, the worst case delay in observing the SFP overflowing is 20 hours.

Assuming the dilution is detected 20 hours after the pool begins to overflow, 13.7 hours

are available to terminate the leakage flow and/or to add boron/borated water to the SFP.

The leakage would be obvious while walking in the vicinity of the SFP, by the SFP high level alarm, which would be initiated (at most) after 66 minutes, and by the unusual demand on the fire protection system—and the pump auto start alarms. The location of the fire protection standpipe is such that the leakage would be immediately evident--in order to reach the SFP it would be a jet of water crossing the normal walk way used to inspect the SFP.

5.2.3 City Water Leakage

The city water line was observed to be 3/4", Ref. [7]. Conservatively assuming the line has a 1" diameter, it is reasonable to assume this source cannot provide flow greater than 50 gpm. As such it is considerably smaller than calculated for the Fire Protection Standpipe failure above.

A leakage of 50 gpm would result in 3.2 days to dilute the SFP from 2000 to 786 ppm boron. Although the this leakage would not be detected by the required boron concentration testing frequency, the leakage is reasonably large and would be easily observed during the required operator rounds which occur every 12 hours, and during normal security rounds which typically occur every couple of hours. The pool water would begin to overflow at 4.2 hours.

Assuming the dilution is detected 20 hours after the pool begins to overflow, 2.2 days are available to terminate the leakage flow and/or to add boron/borated water to the SFP.

The leakage would be obvious in walking in the vicinity of the SFP, by the SFP high level alarm which would be initiated (at most) after 2.4 hours. The location of the city water pipe is such that the leakage would probably be immediately evident even without looking at the pool as the pipe rises up immediately adjacent to the walkway used to inspect the SFP.

5.2.4 Primary Water Pipe Leakage

The normal water supply for SFP makeup is the primary water pipe. Thus, leakage is addressed by the dilution via the primary water make up--see Section 4.1.

5.2.5 SFP Roof: Precipitation (Water) In-Leakage

In order to demonstrate that the effect of any incursion of rain is negligible, the amount of rain that would reduce the boron from 2000 to 786 ppm boron in 7 days is calculated. The basis of 7 days is the SFP boron concentration test frequency—i.e., the weekly sampling of the pool boron concentration would identify the dilution.

The SFP roof area is 4560 sq ft, and the volume of water necessary to dilute the SFP from 2000 to 786 ppm boron is 230,551 gallons--which corresponds to 11.6 inches of rain per day for 7 consecutive days. This is an incredible amount for the Indian Point Unit 2 location. Additionally, the entry into the building of essentially all rain falling on the roof would require significant damage to the roof--which would be unlikely, and readily apparent.

Similarly, in order to accumulate sufficient snow to dilute the pool from 2000 to 786 ppm boron, the roof would need to accumulate a total of about 162 inches of snow (the density of snow with moderate compacting is about 50% of the liquid density)--this snowfall is incredible, and in order to enter the SFP it would require catastrophic failure of the roof--which would not occur with any credible snow accumulation.

6 CONCLUSIONS

An analysis of the potential sources, which can dilute the spent fuel pool has been performed. This analysis demonstrates that sufficient time is available to detect the dilution of the pool, and to mitigate the dilution, prior to the pool boron concentration decreasing from an initial 2000 ppm to a minimum boron concentration of 786 ppm--the value necessary to ensure subcriticality under non-accident conditions.

The analysis characterized the various dilution paths and quantified the rate of water addition and/or the total diluting water volume available. In those cases wherein the rate of water addition is quantified, the following chronology can be calculated from the water addition flow rate: time until SFP high level alarm is reached, time until water overflows the SFP, and time until the boron concentration decreases to the minimum acceptable value. If the diluting water volume is less than 230,551 gallons, the boron concentration cannot decrease from 2000 to 786 ppm.

The dilution events have been categorized as dilution without actual pipe leakage, and dilution associated with a failure of the piping line.

For the dilution without pipeline failure, the worst case is the primary water make-up system. The two primary water pump(s) can draw unborated water from the primary water storage tank and deliver it to the SFP (this system is the principal means of supplying make up water to the SFP); however, the maximum volume of water in the primary water storage tank is insufficient to dilute the SFP from 2000 to 786 ppm boron.

For the dilution due to pipeline failure (excluding failure of the primary water line--which is included in dilution without pipe line failure), the worst case is failure of the 4 inch fire protection standpipe in the SFP building in the vicinity of the SFP. The resulting flow rate (through a .239 square inch crack) is 108 gpm, and the event timetable is as follows:

< 66 minutes: SFP high level alarm
< 116 minutes: SFP water overflows the SFP
> 35.6 hours: SFP boron decreases to 786 ppm

The leakage would be apparent due to the control room alarm associated with the fire header low pressure and fire pump auto-start at time 0, as well as the control room alarm associated with the SFP water alarm.

Subsequently, after a nominal 67 minutes, and no later than 116 minutes, the SFP water would overflow onto the SFP building floor, and then to the pavement exterior to the south end of the SFP building.

The leakage would be obvious to anyone walking in the vicinity of the SFP (and the south end of the building), and by the unusual demand on the fire protection system. The location of the fire protection standpipe is such that the leakage would be immediately evident--in order to reach the SFP it would be a jet of water crossing the normal walk way used to inspect the SFP. In addition, the unusual demand on the fire protection system would alert the operator, who would assist in the termination of the event.

Finally, the termination of the event would be required after almost 1.5 day (35.6 hours) or more, in order to prevent the boron concentration from reaching 786 ppm.

Based on the analysis performed, the credible paths for boron dilution lead to acceptably large times available to detect any dilution of the SFP, and mitigation of the dilution prior to reaching unacceptably low boron concentrations (i.e., below 786 ppm).

7 LIST OF REFERENCES

[1] WCAP-14181, "Westinghouse Owners Group Evaluation of the Potential for Diluting PWR Spent Fuel Pools," Westinghouse Electric Corporation, July 1995

[2] Branch Technical Position SPLB 3-1, Rev. 2, October 1990, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

[3] Branch Technical Position MEB 3-1, Rev. 1, July 1981, Postulated Rupture Locations in Fluid Systems Inside and Outside Containment

[4] NET-173-01, "Criticality Analysis for Soluble Boron and Burnup Credit in the Con Edison Indian Point Unit No. 2 Fuel Storage Racks," Northeast Technology Corp., 12 August 2001

[5] Speyer, Daniel, "IP 2 Rerack Thermal Hydraulic Analysis," Subject No. DMS-04A, 3/8/89

[6] Chicago Bridge & Iron DWG # 20352

[7] U.E & C. DWG. No. 9321-F-2695-13

[8] Con Edison, "Alarm Response Procedure, SGF Window 2-2"

[9] The following constitutes a single composite reference:

- U.E. & C. DWG # 9321-F-2522-11
- U.E. & C. DWG # 9321-F-7029-23
- U.E. & C. DWG # 9321-F-2633-33
- Con Edison Microfiche # 34C636
- Con Edison Microfiche # 34C638