Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

NOV 6 2001

Docket No. 50-336
B18501

RE: 10 CFR 50.90

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

Millstone Power Station, Unit No. 2
Technical Specifications Change Request (TSCR) 2-10-01
Fuel Pool Requirements

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 2. DNC is proposing to change the following Technical Specifications:

- 3.9.16.2 Shielded Cask
- 3.9.17 Movement of Fuel in Spent Fuel Pool
- 3.9.18 Spent Fuel Pool - Reactivity Condition
- Figure 3.9-1A Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C
- Figure 3.9-1B Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C with Poison Pins Installed
- Figure 3.9-2 Spent Fuel Pool Arrangement
- Figure 3.9-4 Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region A
- 3.9.19 Spent Fuel Pool - Storage Pattern
- 5.3.1 Fuel Assemblies
- 5.6.1 Criticality
- 5.6.3 Capacity

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The bases for these Technical Specification Limiting Conditions for Operation (LCO) will also be modified to address the proposed changes.

Rec'd at NRC/DCD 12/17/01

The above Technical Specification Changes implement the following proposed changes:

- Increase the allowable nominal average fuel assembly enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for all regions of the spent fuel pool, the new fuel storage racks (dry), and the reactor core.
- Allow fuel to be located in 40 Region B storage cells which are currently empty and blocked. The cell blockers will be retained, and fuel is proposed to be stored under the cell blockers. The cell blockers still serve a useful function, since the fuel stored in these 40 locations have very restrictive reactivity requirements.
- Credit spent fuel pool soluble boron for reactivity control during normal conditions to maintain spent fuel pool $K_{eff} \leq 0.95$.

There are no physical changes in the plant hardware to implement these changes.

Attachment 1 provides a discussion of the proposed changes and the Safety Summary. Attachment 2 provides the analyses demonstrating the proposed changes do not involve a Significant Hazards Consideration. Attachments 3 and 4 provides marked-up and retyped versions of the current Millstone Unit No. 2 Technical Specifications respectively. Attachment 5 provides the criticality analysis. Attachment 6 provides the boron dilution analysis.

<u>Environmental Considerations</u>

DNC has evaluated the proposed change against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.22. DNC has determined that the proposed changes meet the criteria for categorical exclusion set forth in 10 CFR 51.22(c). The proposed amendment also does not involve irreversible consequences in accordance with 10 CFR 50.92(b).

This determination is based on the fact that the changes are being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to use of a facility component located within the restricted area, as defined by 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment requests meets the following specific criteria:

(i) The proposed change involves no significant hazards consideration.

As demonstrated in Attachment 2, the proposed changes do not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released off site.

The proposed change to Technical Specification 5.3.1 would permit use of fuel assemblies enriched with Uranium 235 increased from 4.5 weight percent to 4.85 weight percent. The safety consideration associated with reactor operation with higher enrichment and extended irradiation have been evaluated by the Nuclear Regulatory Commission (NRC) (February 29, 1988, 53FR60451) and the NRC has concluded that there are no significant adverse radiological or non-radiological impacts associated with the use of higher enrichment and extended irradiation. No changes are being made in the types or amounts of any radiological effluents that may be released offsite during normal operation and design basis accidents.

The environmental impacts of transportation resulting from the use of higher enrichment fuel and extended irradiation have been evaluated by the NRC (July 7, 1988, 53FR30355) and the NRC has concluded that the environmental cost contribution of the proposed increase in fuel enrichment and irradiation limits are either unchanged or may in fact be reduced from those summarized in Table S-4, as set forth in 10 CFR 51.52(c).

The other proposed Technical Specification changes will not change the types or increase in amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the hardware of the facility. There will be no change in the level of controls or methodology used for processing radioactive effluents or handling of solid radioactive waste. There will be no change to the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational exposure resulting from the proposed changes.

<u>Conclusions</u>

The proposed changes do not involve a significant impact on the public health and safety (see the Safety Summary provided in Attachment 1), and do not involve a Significant Hazards Consideration (SHC) pursuant to the provisions of 10 CFR 50.92 (see the evaluation provided in Attachment 2).

<u>Site Operations Review Committee and Nuclear Safety Assessment Board</u>

The Site Operations Review Committee and Nuclear Safety Assessment Board have reviewed and concurred with the determinations.

Schedule

DNC requests issuance of this amendment by July 30, 2003, with the amendment to be implemented within 60 days of issuance. This will allow Millstone Unit No. 2 to use the proposed changes in preparation for and during the Millstone Unit No. 2 refueling outage 15, which is currently scheduled in October of 2003.

State Notification

In accordance with 10 CFR 50.91(b), a copy of this License Amendment Request is being provided to the State of Connecticut.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price, Vice President
Nuclear Technical Services - Millstone

Sworn to and subscribed before me

this 6th day of November, 2001

Sandra J Anton
Notary Public

My Commission expires SANDRA J. ANTON NOTARY PUBLIC COMMISSION EXPIRES MAY 31, 2005

Attachments (6)

cc: H. J. Miller, Region I Administrator
J. T. Harrison, NRC Project Manager, Millstone Unit No. 2
NRC Senior Resident Inspector, Millstone Unit No. 2

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Attachment 1

Millstone Power Station, Unit No. 2

Technical Specifications Change Request 2-10-01
Fuel Pool Requirements
Discussion of the Proposed Changes and Safety Summary

Technical Specification Change Request (TSCR) 2-10-01
Fuel Pool Requirements
Discussion of Proposed Changes and Safety Summary

Introduction

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 2. DNC is proposing to change the following Technical Specifications:

- 3.9.16.2 Shielded cask
- 3.9.17 Movement of Fuel in Spent Fuel Pool
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- Figure 3.9-1A Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C
- Figure 3.9-1B Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C with Poison Pins Installed
- Figure 3.9-2 Spent Fuel Pool Arrangement
- Figure 3.9-4 Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to permit storage in Region A
- 3.9.19 Spent Fuel Pool - Storage Pattern
- 5.3.1 Fuel Assemblies
- 5.6.1 Criticality
- 5.6.3 Capacity

The Bases for these Technical Specification Limiting Conditions for Operation (LCO) will also be modified to address the proposed changes.

The proposed changes to the above Technical Specifications address the following objectives:

- Increase the allowable nominal average fuel assembly enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for all regions of the spent fuel pool, the new fuel storage racks (dry), and the reactor core.
- Allow fuel to be located in 40 Region B storage cells which are currently empty and blocked. The cell blockers will be retained, and fuel is proposed to be stored under the cell blockers. The cell blockers still serve a useful function, since the fuel stored in these 40 locations have very restrictive reactivity requirements.
- Credit spent fuel pool soluble boron for reactivity control during normal conditions to maintain spent fuel pool $K_{eff} \leq 0.95$.
- Reduce Boraflex reactivity credit in Region A and B of the spent fuel pool.

There are no physical changes in the plant hardware to implement these changes.

The reasons for the above proposed changes are:

At present, Millstone 2 is in Cycle 14 operation. The Millstone 2 Spent Fuel Pool will lose the capacity for a full offload of the reactor core at the end of cycle 15. This proposed change recovers the use of the 40 currently blocked cells for the storage of spent fuel. If fuel can be located in the 40 cells currently blocked in Region B, the loss of Full Core Reserve will be delayed until end of cycle 16.

Increasing the allowable nominal average fuel assembly enrichment from 4.5 w/o to 4.85 w/o U-235 for all spent fuel pool regions, the new fuel storage vault, and the reactor core will allow for more flexibility in cycle length designs. Also, it will potentially reduce the amount of spent fuel generated by allowing the potential for reduced feed fuel batch sizes.

Credit for soluble boron for reactivity control under normal conditions is primarily needed to offset the reduced boraflex reactivity credit.

Reduced boraflex reactivity credit in Region A and B is being implemented to make allowance for possible future degradation of the boraflex. By reducing the amount of boraflex reactivity credit taken, this allows time to respond should boraflex degradation be detected by the in-service testing program.

Current Millstone 2 Spent Fuel Pool Configuration

The Millstone 2 spent fuel pool consists of 3 regions of spent fuel storage racks, designated Regions A, B and C. TS Figure 3.9-2 shows a schematic of the pool layout. The Region A and B racks contain boraflex as the active neutron absorber in a flux trap design. The Region C racks are an eggcrate design with no fixed neutron absorber. Fuel may be stored in three (3) types of configurations in Region C per existing Technical Specifications. Fuel assemblies stored in Region C may be stored with or without borated stainless steel rodlets (for reactivity control), and Consolidated Fuel Storage Boxes (CFSB) are also allowed to be stored in Region C. In a letter dated June 2, 1987,[1] the Nuclear Regulatory Commission (NRC) approved changes to the Technical Specifications which allow storage of CFSB in the Spent Fuel Pool. A CFSB contains the fuel rods from 2 fuel assemblies stored in a tight matrix. The CFSB has essentially the same dimensional envelope as a fuel assembly. The only storage cells which are prohibited by Technical Specifications from storing fuel in the spent fuel pool are 40 storage locations in Region B, which are empty and blocked for reactivity control.

Soluble boron is currently credited in the spent fuel pool for reactivity control only for accident conditions.

The maximum allowed nominal average fuel assembly enrichment is 4.5 w/o U-235 for the spent fuel pool, new fuel storage (dry) racks and reactor core.

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(1) D. H. Jaffe (USNRC) letter to E. J. Mroczka, "Issuance of Amendment 117," dated June 2, 1987.

Technical Specification Changes

LCO 3.9.16.2

LCO 3.9.16.2 is proposed to be entirely deleted. This LCO related to the required spent fuel pool soluble boron concentration for shielded cask movement. This LCO will be merged with proposed LCO 3.9.17, "Spent Fuel Pool Boron Concentration." In addition the bases for LCO 3.9.16 have been changed to reflect the deletion of LCO 3.9.16.2.

LCO 3.9.17

LCO 3.9.17 currently is titled "Movement of Fuel in Spent Fuel Pool." The existing LCO requires 800 ppm of soluble boron in the spent fuel pool water whenever a fuel assembly or a Consolidated Fuel Storage Box (CFSB) is moved.

The title of proposed LCO 3.9.17 is changed to "Spent Fuel Pool Boron Concentration". The LCO will require that the spent fuel pool (SFP) soluble boron concentration be $\geq$ 1720 ppm, with an APPLICABILITY of whenever any fuel assembly or a CFSB is stored in the SFP.

It is important to recognize that the existing LCO 3.9.17 requires SFP boron concentration to be verified only when fuel is moved. The proposed LCO 3.9.17 is applicable whenever fuel is stored in the SFP.

As described later in the safety summary, the criticality analysis demonstrates that 1400 ppm of soluble boron is sufficient to provide enough negative reactivity to ensure that accident conditions will not cause K_{eff} of the SFP to exceed 0.95. Design basis accident conditions for which soluble boron credit is required are, a dropped or misplaced fuel assembly, a dropped or misplaced CFSB, or a shielded cask drop onto the storage racks. The chosen LCO value of 1720 ppm of soluble boron exceeds 1400 ppm and provides safety margin. The proposed APPLICABILITY statement does not need to mention a shielded cask, since the applicability is whenever any fuel assembly or a CSFB is stored in the SFP. If there was no fuel or CSFB stored in the SFP, a shielded cask drop would have no reactivity consequences.

The proposed ACTION statement, should the soluble boron concentration fall below 1720 ppm, is to suspend the movement of all fuel, CFSBs and shielded casks, and immediately initiate action to restore the SFP soluble boron concentration to within its limit. This action ensures that accident conditions such as a fuel assembly/CFSB drop, a fuel assembly/CFSB misplacement, or a shielded cask drop into the SFP, are prevented from occurring, since these accident conditions credit soluble boron.

A statement that LCO 3.0.3 is not applicable has been added since there is no relation between the spent fuel pool ACTION requirements and plant power operation.

The proposed LCO 3.9.17 SURVEILLANCE requirements are the same as the current LCO 3.9.17 SURVEILLANCE requirements except for:

- the new surveillance requirement changes the required soluble boron concentration from 800 ppm to 1720 ppm to be consistent with the revised criticality analysis,
- the new surveillance requirement adds "shielded cask over the cask laydown area" as a condition for performing the surveillance, since existing LCO 3.9.16.2 was merged with this proposed LCO.
- The proposed surveillance interval is every 7 days. This proposed surveillance is performed whenever fuel is stored in the SFP, and there is no corresponding existing surveillance whenever fuel is stored in the SFP. Therefore, this is a more conservative surveillance requirement then currently exists.

In addition the bases for LCO 3.9.17 have been changed to reflect the above changes.

LCO 3.9.18

LCO 3.9.18 currently is titled "Spent Fuel Pool - Reactivity Condition". The proposed LCO 3.9.18 is titled " Spent Fuel Pool - Storage". This LCO has been extensively re-written to be consistent with Improved CE STS (NUREG 1432 Revision 2) LCO 3.7.18. The proposed LCO 3.9.18 is changed only for clarification purposes, and no changes were necessary to support the underlying design changes proposed by this license amendment request. No bases changes are made other than to change the title of the LCO. These changes are meant to be TS improvements, and are not directly connected with the proposed changes.

TS Figure 3.9-1A

This TS figure shows the minimum required fuel assembly exposure as a function of initial enrichment to permit storage of fuel assemblies in Region C. This TS figure is revised to reflect the revised criticality analysis.

The values shown in this Figure are taken from the criticality analysis that is attached. The required burnups shown in the proposed Figure 3.9-1A are slightly smaller than the current limits due to soluble boron credit.

TS Figure 3.9-1B

This TS figure shows the minimum required fuel assembly exposure as a function of initial enrichment to permit storage of fuel assemblies in Region C, with poison pins installed. This TS figure is revised to reflect the revised criticality analysis.

The values shown in this Figure are taken from the criticality analysis that is attached. The required burnups shown in the proposed Figure 3.9-1B are slightly smaller than the current limits due to soluble boron credit.

TS Figure 3.9-2

This proposed TS figure is revised only to provide a clearer Figure to show the spent fuel pool arrangement. There are no changes to the Figure.

TS Figure 3.9-4

This TS Figure shows the minimum required fuel assembly exposure as a function of initial enrichment to permit storage of fuel assemblies in Region A. This TS figure is revised to reflect the revised criticality analysis.

The values shown in this Figure are taken from the criticality analysis that is attached. The required burnups shown in the proposed Figure 3.9-4 are slightly smaller than the current limits due to soluble boron credit.

LCO 3.9.19

LCO 3.9.19 is revised to reflect that Batch B fuel assemblies are now allowed in the 40 Region B storage locations that have cell blockers.

3.9.19(1) is revised to add a sentence acknowledging that a Batch B fuel assembly may be stored underneath the cell blocker. 3.9.19(2) is revised to state that if a cell blocker is removed, all cells except the location with the removed cell blocker device must be vacant of fuel. This acknowledges that storage of a fuel assembly in the cell blocker location is now possible.

A statement that LCO 3.0.3 is not applicable has been added since there is no relation between the spent fuel pool ACTION requirements and plant power operation.

A footnote is added to clarify that a Batch B fuel assembly refers to any of the Batch B fuel assemblies which were part of the first Millstone 2 core. All Batch B fuel assemblies meet the requirements of the attached criticality analysis, so that all Batch B fuel assemblies are interchangeable for being stored in Region B cell blocker locations.

A footnote is also added to provide an exception to LCO 3.9.19 during the initial installation of Batch B fuel assemblies in the cell blocker locations. The reasons and justification of this exception are discussed in the safety summary. The term "initial installation" refers to a period of time shortly after NRC approval of this license amendment, when the 40 Batch B fuel assemblies are initially placed into the cell blocker locations.

In addition, the bases for LCO 3.9.19 have been changed to reflect the capability to store fuel under the cell blockers, and the justification for the exception to the LCO during initial installation of the Batch B fuel under the cell blockers.

Design Feature 5.3.1

Design Feature 5.3.1 is revised to change the allowed reactor core fuel assembly maximum nominal average enrichment from 4.5 w/o U-235 to 4.85 w/o U-235. The addition of the word "nominal average" here and in other Design Feature sections provides consistency with the analytical approach used, and is bounded by the criticality analyses provided here. The addition of the sentence stating that the maximum fuel rod enrichment is 5.0 w/o U-235 acknowledges an NRC requirement stated in their SER[(2)] for WCAP 14416-NP-A.

Design Feature 5.6.1

Design Feature 5.6.1a) is revised to change the allowed fuel assembly maximum nominal average enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for the new fuel (dry) storage racks. The addition of the sentence stating that the maximum fuel rod enrichment is 5.0 w/o U-235 acknowledges an NRC requirement stated in their SER.[(2)]

Proposed Design Features 5.6.1b) through 5.6.1h) replace existing Design Features 5.6.1b) through 5.6.1e). The wording format of these proposed design features is intended to comply with the NRC SER[(2)] contained in WCAP-14416-NP-A. These proposed design feature changes reflect the following changes:

- Fuel assembly maximum nominal average enrichment increased from 4.5 w/o U-235 to 4.85 w/o U-235.
- Reflect credit for soluble boron in the spent fuel pool.

Design Feature 5.6.3

Design Feature 5.6.3 is revised to delete the existing footnote. This change reflects the ability to store 40 fuel assemblies in the Region B locations that have cell blockers.

Safety Summary

Proposed Changes by this License Amendment

The proposed Technical Specification Changes addresses the following proposed changes:

- Increase the allowable nominal average fuel enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for all regions of the spent fuel pool, the new fuel storage racks (dry), and the reactor core.

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(2) T. E. Collins (USNRC) letter to T. Greene (Westinghouse Owners Group) "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P," Westinghouse Spent Fuel Rack Criticality Analysis Methodology (TAC NO. M93284)," dated October 25, 1996.

- Allow irradiated fuel to be located in 40 Region B storage cells which are currently empty and blocked. The cell blockers will be retained, and fuel is proposed to be stored under the cell blockers. The cell blockers still serve a useful function, since the fuel stored in these 40 locations have very restrictive reactivity requirements.
- Credit spent fuel pool soluble boron for reactivity control during normal conditions to maintain spent fuel pool $K_{eff} \leq 0.95$.
- Reduce Boraflex reactivity credit in Region A and B of the spent fuel pool.

Safety Considerations

DNC addressed the following issues as the most significant safety considerations for the proposed changes.

In allowing an increase in fuel enrichment from a nominal average fuel assembly enrichment of 4.5 w/o U-235 to 4.85 w/o U-235, all regions of the spent fuel pool, the fuel transfer equipment in the transfer canal, the new fuel storage racks (dry), and the reactor core must be considered to ensure that the criticality analysis bounds the use of higher enrichment fuel. The spent fuel pool criticality analyses are intended to comply with the requirements specified in WCAP-14416-NP-A and the NRC SER contained therein. Where fuel burnup is credited, conservative approaches are taken for axial burnup effects and reactivity equivalencing issues, to reflect recent NRC concerns in these areas.

Because Boraflex is still credited for reactivity control, although at reduced levels by the proposed analysis, assurance must be provided that the material is capable of continuing to perform its design function.

The increase in fuel storage of 40 additional storage locations, by allowing 40 Batch B fuel assemblies to be stored in the Region B racks, must be within the capability of the racks, and within the capability of the spent fuel pool bulk cooling analysis.

The 40 existing cell blockers will be retained to provide the same level of administrative control that currently exists.

Since a spent fuel pool soluble boron concentration of 600 ppm is credited for reactivity control under normal conditions, assurance must be provided that a spent fuel pool soluble boron dilution event will not cause spent fuel pool boron concentration to be decreased from the LCO minimum value of $\geq$ 1720 ppm, to < 600 ppm.

These design changes do not result in any hardware changes to the plant. The cell blockers are already of a removable design. There are no modifications necessary to the cell blockers to allow storage of fuel under the cell blockers. There are no changes in how the stainless steel poison pins are used in Region C of the spent fuel pool. There are no changes in how fuel is moved, or the process used to qualify and verify fuel storage in the pool.

From an operational perspective, the proposed design changes are transparent. Spent fuel pool soluble boron concentration has always been maintained at levels in excess of the proposed minimum 1720 ppm LCO requirement. Therefore, there is no practical change; the proposed changes take a partial reactivity credit, for soluble boron presently in place. While the enrichment versus burnup values are changing on proposed TS Figures 3.9-1A, 3.9-1B and 3.9-4, there are no changes in how fuel is moved, or in any method of how administrative controls are used to ensure that fuel is not misloaded.

Spent Fuel Pool Criticality Analysis - General

The criticality analysis to support the proposed changes was performed by Westinghouse. A copy of the analysis is provided in Attachment 5. The analysis was performed in accordance with the criteria of WCAP-14416-NP-A. The objectives, design criteria and methodology are described in Section 1 of Attachment 5.

The NRC's concerns expressed in the letter dated July 27, 2001,[(3)] are addressed as described in the attached criticality analysis. As requested in this NRC letter, the proposed Technical Specification 5.6.1 does not reference WCAP-14416-NP-A. As detailed in the attached criticality report, axial burnup effects were explicitly modeled and very conservative axial burnup distributions were taken to bound this effect.

The NRC has also expressed concern, as documented in NUREG-6683, that reactivity equivalencing used to model fuel burnup using an equivalent fresh enrichment, could be used in a non-conservative manner. These concerns are also addressed, as documented in Attachment 5, by use of the DIT computer code generated burned fuel isotopics, which are then used in the KENO computer code to explicitly model the burned fuel. Therefore, fresh fuel equivalent enrichments are not used.

This criticality analysis assumes that the maximum reactivity fuel assembly is a fresh (no burnup) fuel assembly, with an assembly average enrichment of 4.85 w/o U-235. This criticality analysis has been conservatively performed by not crediting any integral fuel burnable absorbers, which are typically present in fresh fuel, and with no credit for grids. The most reactive fuel design is used for each storage region, and the most reactive spent fuel pool water temperature is used for each region. A tolerance and uncertainty analysis is also provided.

Millstone 2 style fuel assemblies typically have different fuel pin enrichments in the radial direction. While the nominal assembly average enrichment limit is proposed to be 4.85 w/o U-235, individual fuel pins may have enrichments as high as 5.0 w/o U-235. The criticality report provided in Attachment 5 models the fuel assembly with all fuel pins at the average radial enrichment. Modeling of an average radial fuel pin

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(3) S. Dembek (USNRC) letter to H. A. Sepp (WEC), "Non-Conservatism in Axial Burnup Biases for Spent Fuel Rack Criticality Analysis Methodology," dated July 27, 2001.

enrichment is conservative relative to modeling the distributed radial pin enrichments. It is also possible that reduced enrichment axial blankets may be used. The attached criticality analysis bounds the use of reduced enrichment axial blankets.

Spent Fuel Pool Criticality Analysis - Boraflex Modeling

The attached criticality analysis takes less reactivity credit for boraflex than is currently credited. The Region A and Region B spent fuel racks contain boraflex as the active neutron absorber.

The existing analyses of record for Region A and B credit boraflex with the original design areal density of 0.033 +/- 0.003 grams B-10/cm2. The existing analysis also assumes that 5.65" gaps exist in all boraflex panels, with a random axial distribution of the gaps.

The attached criticality analysis reduces the reactivity credit for boraflex in both Region A and B by crediting a minimum areal density of .025 grams B-10/cm2. This is equivalent to stating that approximately a 25% reduction in boraflex thickness would have to occur before impacting the criticality analysis. Also the boraflex gap model has been made significantly more conservative by changing how the gaps are axially distributed. The attached criticality analysis is similar to the existing criticality analysis, in that it conservatively penalizes that 5.65" gaps exist in all boraflex panels. The difference is that in this revised criticality analysis, the boraflex gaps are all lined up in 2 axial elevations, near the fuel centerline. This causes significantly more reactivity insertion due to the gap locations.

As will be described later in more detail, the boraflex at Millstone 2 is currently performing in an adequate manner. The above boraflex modeling changes are being made to provide margin in case future in-service testing of the boraflex indicates unexpected degradation.

Spent Fuel Pool Criticality Analysis - Normal Storage Conditions

Region A of the spent fuel storage pool is designed to ensure a $k_{eff} \leq 0.95$ with the storage pool filled with water borated to a minimum concentration of 600 ppm for normal conditions. Fuel assemblies stored in this region must comply with the burnup and initial enrichment limit in the proposed TS Figure 3.9-4 to ensure that the proper burnup has been sustained. Burnups in proposed figure 3.9-4 refer to average assembly fuel burnups. Enrichments in proposed figure 3.9-4 refer to the average assembly enrichment. If axial blankets are present, then the center zone average enrichment would be used.

Region B of the spent fuel storage pool is designed to ensure a $k_{eff} \leq 0.95$ with the storage pool filled with water borated to a minimum concentration of 600 ppm for normal conditions. Fresh fuel assemblies stored in this region may have a maximum nominal average enrichment of 4.85 weight percent U-235. These fuel assemblies are placed in a 3 out of 4 configuration for reactivity control.

For Region B, to ensure that unqualified fuel assemblies are not placed in the 4th location, cell blockers are used in the 4th location. As determined by analysis, the 4th location may either remain empty or be filled by a low reactivity fuel assembly. The low reactivity fuel assembly is a Batch B Combustion Engineering designed fuel assembly, which has a maximum initial enrichment of 2.36 weight percent, and a minimum burnup of 22,300 mwd/mtu. All Batch B fuel assemblies meet this requirement. The existing cell blocking devices will be retained in the 4th location, except when inserting or removing a low reactivity (Batch B) fuel assembly.

Region C of the spent fuel storage pool is designed to ensure a $k_{eff} \leq 0.95$ with the storage pool filled with water borated to a minimum level of 600 ppm for normal conditions. Fuel assemblies stored in this region must comply with proposed TS Figures 3.9-1A or 3.9-1B to ensure that the proper burn-up has been sustained. Additionally, fuel assemblies utilizing proposed TS Figure 3.9-1B require that borated stainless steel poison pins are installed in the fuel assembly's center guide tube and in two diagonally opposite guide tubes. The poison pins are solid 0.87 inch O.D. borated stainless steel, with a boron content of 2 weight percent boron. The use of these poison pins has been previously approved by the NRC, and there are no changes to how the poison pins are utilized. Burnups in proposed figure 3.9-1A and 3.9-1B refer to average assembly fuel burnups. Enrichments in proposed figure 3.9-1A and 3.9-1B refer to the average assembly enrichment. If axial blankets are present, then the center zone average enrichment would be used.

Region C of the spent fuel storage pool is also currently designed to permit storage of consolidated fuel storage boxes (CFSBs), such that $k_{eff} \leq 0.95$ is maintained with the storage pool filled with unborated water. The contents of the CFSB's to be stored in Region C must comply with existing TS Figure 3.9-3. DNC does not see any need to change this figure, since we do not anticipate the need to consolidate fuel of an enrichment > 4.5 w/o U-235, nor is there any need to lower the burnup requirements for CFSBs by crediting soluble boron. Therefore, the existing TS Figure 3.9-3 will be retained. This figure conservatively takes no credit for soluble boron for normal conditions, therefore retaining this curve is conservative. Therefore there are no changes with regard to storage of CFSBs in the spent fuel pool, and no additional analysis has been performed for storage of CFSBs.

The above analyses show that for all Regions of the SFP 600 ppm of soluble boron is needed under normal conditions to assure not exceeding a spent fuel pool k_{eff} of 0.95 (including biases and uncertainties). Also, the criticality analysis shows that even with 0 ppm of soluble boron, under normal conditions, spent fuel pool k_{eff} will be less than 1.00 (including biases and uncertainties).

Spent Fuel Pool Criticality Analysis - Accident Conditions

The spent fuel pool criticality analysis has analyzed the following accident conditions:

- a single 4.85 w/o U-235 fresh fuel assembly is misloaded in Region A, B or C.
- a single 4.85 w/o U-235 fresh fuel assembly is misloaded between the new fuel elevator and Region C.
- a single 4.85 w/o U-235 fresh fuel assembly is misloaded outside of the region A and C racks.
- a single 4.85 w/o U-235 fresh fuel assembly is dropped on top of the fuel racks.
- A heavy load is dropped into Regions A, B or C.
- SFP bulk water temperature exceeds 150F.

For these accident conditions, credit for soluble boron is acceptable per the double contingency principle.

Based on the criticality analysis provided in Attachment 5, the limiting accident is a heavy load (shielded cask) drop onto the Region A racks, which would require an additional 800 ppm of soluble boron. The total amount of soluble boron required would be the 800 ppm to compensate for the reactivity increase from the heavy load drop, plus the 600 ppm for normal conditions, for a total of 1400 ppm. This value of 1400 ppm ensures that K_{eff} is ≤ 0.95 including all uncertainties and biases for the heavy load drop, and bounds all other less limiting accidents.

The proposed TS will require 1720 ppm of soluble boron at all times fuel is in the spent fuel pool.

Criticality Analysis - New Fuel Storage Vault and Transfer Machine

Increasing the fuel assembly enrichment limit from 4.5 w/o to 4.85 w/o U-235 also requires ensuring that the new fuel storage vault (dry) and the spent fuel pool transfer carriage are capable of storing new fuel of this increased enrichment. The spent fuel pool fuel transfer carriage moves fuel back and forth from Containment to the spent fuel pool. In a letter dated April 10, 1990,[(4)] Technical Specification changes were submitted to the NRC to allow enrichments up to 4.5 weight percent U-235 to be stored in the new fuel storage racks and to allow 4.5 weight percent U-235 to be the maximum fuel enrichment in the reactor core. The analysis, performed by Advance Nuclear Fuels Corporation (ANF), evaluated the loading of fuel of enrichments up to 5.0 weight percent into the new fuel storage racks and the SFP transfer carriage. In a letter dated June 13, 1990,[(5)] the NRC approved changes to the Technical Specification 5.6.1(a) to allow enrichments up to 4.5 weight percent U-235 to be stored in the new fuel storage

(4) E. J. Mroczka letter to the U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Proposed Change to Technical Specifications Fuel Enrichment Limits," dated April 10, 1990.

(5) G. S. Vissing (USNRC) letter to E. J. Mroczka, "Issuance of Amendment 146 (TAC NO. 76473)," dated June 13, 1990.

racks and Technical Specification 5.3.1 to allow 4.5 weight percent U-235 to be the maximum fuel enrichment in the reactor core. The same ANF analysis is still valid for fuel enrichments up to 5.0 weight percent U-235, which is still conservative relative to the proposed enrichment increase to 4.85 weight percent U-235.

Use of 4.85 w/o fuel in the Reactor Core

Use of fuel assemblies with an average enrichment up to 4.85 w/o U-235 in the reactor core is handled as part of the normal process for a reload design. Each reload design is analyzed by the fuel vendor and/or licensee under the 10 CFR 50.59 process to determine that the reload design meets the required safety criteria. Also, increasing the fuel enrichment limit to 4.85 w/o U-235 by itself has no effect on the maximum allowed fuel burnup limits, which are not changing.

Increased Fuel Storage

The spent fuel pool fuel storage capacity will be increased by 40 locations with the implementation of the proposed changes.

Mechanical and Seismic analyses for the fuel storage racks were approved by the NRC in TS Amendments 109[(6)] and 117.[(1)] The fuel storage racks were designed for storage of fuel (including consolidated fuel) in all storage locations. As a result, while the Region B fuel storage racks under this design change will store 40 additional fuel assemblies, that is within the original design of the racks, as approved by the NRC. The Region A and B racks were licensed by the NRC for storage of fuel in all Region A and B storage locations, up until TS amendment 158 in 1992. Amendment 158[(7)] installed the 40 cell blockers only because additional reactivity controls were needed under the storage configuration at that time. The criticality analysis provided here, shows that fuel may be stored under the 40 cell blockers. In summary, the original rack mechanical and seismic analyses approved by the NRC in TS amendments 109[(6)] and 117[(1)] are still valid, and bounds the use of all Region B locations for fuel storage.

The current design basis heat load analysis for the Millstone 2 spent fuel pool was revised in accordance with 10 CFR 50.59. This heat load analysis already conservatively accounts for storage of 40 fuel assemblies under the cell blockers. The SFP heat load analysis of record maximizes heat load by having all SFP storage locations filled with fuel at the end of plant life. The analysis conservatively assumes the storage locations are filled by the most recently discharged fuel, which will result in the largest heat load. The Batch B fuel assemblies have a decay time > 16 years, which is a longer decay time than the decay time of any fuel in the heat load analysis of record. Thus the Batch B fuel decay heat load is bounded by the analysis of record. Therefore, while 40 additional fuel storage locations are now available per this

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(6) D. B. Osborne (USNRC) letter to J. F. Opeka, "Issuance of Amendment 109," dated January 15, 1986.

(7) G. S. Vissing (USNRC) letter to J. F. Opeka, "Issuance of Amendment 158 (TAC No. M83180)," dated June 4, 1992.

proposed change, the heat load analysis of record already bounds storage of fuel assemblies under the cell blockers.

Storage of fuel under the cell blockers does not cause any inability to cool those 40 assemblies with the cell blockers covering those locations. The cell blockers are of an open design where the blocked area is not sufficient to restrict flow and inhibit heat removal.

Calculations also have been performed which show that the increased enrichment limit from 4.5 w/o to 4.85 w/o U-235 will not increase the previously calculated decay heat loads.

Boron Dilution Analysis

Because soluble boron in the SFP is now credited for normal conditions, the possibility of a SFP boron dilution accident must be considered. Attached to this proposed TS change as Attachment 6 is the summary of the engineering investigation into possible boron dilution scenarios.

The minimum TS SFP soluble boron limit is proposed to be 1720 ppm. Typical SFP soluble boron concentrations are about 2100 ppm. Criticality analysis (Attachment 5) has shown that 600 ppm of soluble boron is needed under normal conditions in the SFP to assure compliance with the 0.95 k_{eff} design basis (including biases and uncertainties). Further, the criticality analysis has shown that even with 0 ppm of soluble boron, under normal conditions in the SFP, the SFP would remain subcritical, ($k_{eff} < 1.0$), including biases and uncertainties.

A TS required spent fuel pool soluble boron concentration of 1720 ppm was selected for the following reasons:

- It is sufficiently high to provide assurance that a postulated boron dilution event can be detected in sufficient time to detect and stop the dilution, prior to reaching 600 ppm.
- The value of 1720 ppm is consistent with existing TS requirement 3.9.1, which requires in Mode 6 that the reactor vessel and refueling canal be maintained $\geq$ 1720 ppm. Thus when the transfer tube is open and the spent fuel pool is connected to the refuel pool, there will be consistent soluble boron requirements.

The engineering analysis (Attachment 6) of potential scenarios which could dilute the boron concentration in the SFP demonstrates that sufficient time is available to detect and mitigate a boron dilution prior to reaching 600 ppm. This assures not exceeding the 0.95 k_{eff} design basis (including biases and uncertainties). It should be noted that for accident conditions, up to 1400 ppm soluble boron is credited in the criticality analysis. However, consideration of a simultaneous occurrence of 2 unlikely and independent events, such as a boron dilution event and another independent accident condition, is not required to be considered by the double contingency principle.

The systems which could dilute the spent fuel pool, either by direct connection to the spent fuel pool, or by a potential pipe crack/break have been analyzed and are discussed in Attachment 6.

The volume of unborated water needed to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm has been calculated to be 230,971 gallons. This calculation uses a continuous feed and bleed dilution, assuming uniform mixing, with very conservative assumptions to minimize the amount of water assumed to be present in the spent fuel pool. Most of the potential dilution sources described in Attachment 6 do not have volumes as large as 230,971 gallons, and therefore are not capable of diluting the SFP boron concentration from 1720 ppm to 600 ppm.

Some systems do have the potential to add in excess of 230,971 gallons of water to the SFP. A dilution flow rate of 200 gpm of unborated water would be necessary for 19 hours to cause a dilution of 230,971 gallons, resulting in a SFP soluble boron concentration change from 1720 ppm to 600 ppm. There is no automatic spent fuel pool level control system in the spent fuel pool, so that any unborated water added to the spent fuel pool will cause a SFP level increase. The operators will be alerted to this dilution by a high SFP water level alarm in the control room, and/or Plant Equipment Operator (PEO) rounds of the SFP general area, detecting high SFP water levels and/or SFP overflow. The conservatively longest time between PEO rounds of the SFP area is 12 hours. As discussed above, for a dilution flowrate of 200 gpm, 19 hours are needed to dilute the SFP soluble boron concentration to 600 ppm, so there is ample time to detect and terminate the dilution event. Attachment 6 shows that a review of Millstone 2 specific systems results in a largest possible dilution flow rate of 100 gpm, for any system with sufficient volume to dilute the SFP to 600 ppm. Therefore there is a wide margin between the largest dilution flowrate of 100 gpm, and the 200 gpm dilution flowrate which corresponds to 19 hours to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm. Therefore, there is at least 19 hours available to detect and terminate a Millstone 2 SFP boron dilution prior to the boron concentration reaching 600 ppm.

In summary, the ability to prevent the SFP soluble boron concentration from being diluted from the TS minimum value of 1720 ppm to a value of 600 ppm has been demonstrated by showing that each potential dilution source meets one of the following two criteria:

- Dilution sources not capable of supplying 230,971 gallons of unborated water will not be capable of diluting the SFP soluble boron concentration from 1720 ppm to 600 ppm.
- For Millstone 2 systems with sufficient volume to dilute the SFP soluble boron concentration to 600 ppm, the largest possible dilution flow rate is 100 gpm. If the dilution flow rate of unborated water is $\leq$ 200 gpm, then at least 19 hours will be needed for the SFP soluble boron concentration to be reduced from 1720 ppm to 600 ppm. All dilution scenarios evaluated here will eventually cause a SFP high water level alarm in the control room, or will be detected by the Plant Equipment Operator (PEO) detecting a high SFP water level and/or SFP overflow. Since the

conservatively longest time between PEO rounds is 12 hours, and 19 hours are needed at 200 gpm to dilute the SFP soluble boron concentration to 600 ppm, there is ample time to detect and terminate the dilution event.

Based on the evaluation in Attachment 6, an unplanned or inadvertent dilution event which would reduce the boron concentration from 1720 ppm to 600 ppm is not credible for MP2. The large volume of water required to dilute the SFP, the TS controls on SFP boron concentration, PEO rounds as well as engineered alarms, would effectively detect a dilution event prior to k_{eff} reaching 0.95. There is at least 19 hours available to detect and terminate a Millstone 2 SFP boron dilution prior to the boron concentration reaching 600 ppm.

It is also important to note that the boron dilution calculations provided in Attachment 6 assume complete mixing of the SFP soluble boron concentration during the dilution. Should the SFP soluble boron concentration reach 0 ppm locally around the fuel due to incomplete mixing, the SFP will remain subcritical, (k_{eff} < 1.0), including all biases and uncertainties.

Another mechanism that is considered in ensuring that SFP soluble boron concentration is maintained ≥ 600 ppm, is the possibility of loss of SFP water level due to a drain down event. The loss of water level itself will not cause a reduction in SFP boron concentration, but the subsequent recover of SFP level could cause a reduction in the SFP boron concentration if un-borated makeup water is used. There are no credible events initiated in the SFP that can cause any significant loss in SFP water level, and therefore any make-up to the SFP even with un-borated water would not significantly reduce the SFP soluble boron concentration. There are however, events during refueling operations, such as a nozzle dam failure or refuel pool seal failure, which could cause significant loss of SFP level, due to the transfer tube being open to containment. If the refueling pool is flooded up and the transfer tube is open between containment and the spent fuel pool, any draindown event that occurs in containment will affect SFP level since the transfer tube is open. It is possible that the addition of un-borated water during recovery from these hypothetical events could cause significant reductions in SFP boron concentration. Procedural controls will be implemented to the existing Abnormal Operating Procedures for these events, to ensure that during the recovery from these events, that soluble boron concentration is maintained > 600 ppm.

Spent Fuel Pool Soluble Boron Concentration Surveillance Requirements

A proposed surveillance interval for monitoring the spent fuel pool soluble boron concentration has been added. Proposed LCO 3.9.17 is that the spent fuel pool soluble boron concentration be monitored every 7 days. Since the applicability of proposed LCO 3.9.17 is now "Whenever fuel is stored in the spent fuel pool," this means that spent fuel pool soluble boron concentration will be determined every 7 days.

A basis for TS LCO 3.9.17 is to ensure that sufficient soluble boron is present to mitigate the reactivity consequences of potential accident conditions such as a dropped or misplaced fuel assembly, CFSB or dropped shielded cask. A 7 day surveillance interval for soluble boron concentration is acceptable for the following reasons:

- No deliberate major replenishment of pool water is expected to take place over this short period of time (7 days). Also, there is a large buffer between the minimum TS soluble boron concentration limit of 1720 ppm, and the required 1400 ppm soluble boron concentration needed for accident conditions.
- TS continue to require verification of the spent fuel pool boron concentration within 24 hours of fuel assembly movement, consolidated fuel movement, or cask movement over the cask laydown area. This verifies that the boron concentration is within limits just prior to the movement.
- Any inadvertent boron dilution would then have to occur essentially concurrent to the fuel/cask movement. The hypothetical boron dilution event is independent of the fuel/cask movement and is not required to be considered by the double contingency principle. Further, it is not credible that while personnel are present in the spent fuel pool for fuel/cask movement, that they would fail to notice SFP level increasing, an overflowing SFP, and/or significant amounts of water entering the spent fuel pool causing the dilution.

An additional basis for TS LCO 3.9.17 is to ensure that sufficient soluble boron is present ($\geq$ 1720 ppm) as a precondition to a potential boron dilution event. A soluble boron concentration of 600 ppm is credited to ensure a SFP $K_{eff} \leq 0.95$ under normal design basis conditions. The potential for dilution of the spent fuel pool soluble boron concentration from 1720 ppm to 600 ppm was discussed previously. It was determined that an unplanned or inadvertent dilution event which would reduce the soluble boron concentration from 1720 ppm to 600 ppm is not credible for Millstone Unit 2.

<u>Boraflex Material Condition</u>

The existing spent fuel racks were installed in 1986. Thus the boraflex in the Millstone 2 spent fuel storage racks has been exposed to water and radiation for about 15 years. There are 384 storage locations which have boraflex, with 4 panels per location, for a total of 384 x 4 = 1536 boraflex panels. The boraflex is initially manufactured to be .11 inch thick, 8.063 inch wide and 141.25 inch long. The boraflex is contained inside a poison box which is stored inside each storage cell, with the poison box itself removable from the storage cell. The boraflex is sandwiched between stainless steel protective sheets.

The boraflex in-service testing program at this time consists of 3 principal parts.

- The first part of the program is direct examination and testing of actual in-service boraflex material. This is accomplished by removing a boraflex poison box from the spent fuel pool, cutting away the stainless steel protective layer, examining the boraflex, and then testing selected portions of the boraflex material. A poison box was so removed in the year 2000. The results of this visual examination and material testing showed that the material was in good condition. Neutron attenuation testing of samples of this in-service boraflex showed no detectable loss of B-10 density. Thus DNC concludes that to date, there has been no detectable loss of B-10 density from the as manufactured condition.

- The second part of the program is blackness testing. Three (3) blackness test campaigns have been performed on selected boraflex panels in the Millstone 2 spent fuel racks. The last such test was performed in 1996, with the following results:

  89 cells (356 panels) of the total of 384 boraflex cells were tested. 64 cells had measurable gaps and 25 cells had no detectable gaps. Of the 64 cells with gaps:

  134 total gaps were measured
  83 of the 134 gaps were measured to be less than 1.0 inch.
  41 of the 134 gaps were measured to be in the range of 1.0 to 1.5 inches
  10 of the 134 gaps were measured to be in the range 1.6 to 1.9 inches

  Distribution of Boraflex gaps was generally random in the axial direction. The largest individual gap found was 1.9 inches. A few boraflex panels had 2 gaps in the panel, with the largest gap being a sum of 2.8" of total gap in that panel. This is far below the 5.65" gap assumed in the criticality analysis.

  Thus blackness testing to date has shown that while gaps are present, they are far less severe than assumed in the criticality analysis. The assumed criticality analysis value of 5.65" boraflex gaps is based on the EPRI limiting value of 4% boraflex shrinkage. Since the original boraflex length is 141.25 inches, the resulting maximum axial gap would be: 141.25 inches x .04 = 5.65 inches.

- The third part of the program is spent fuel pool silica monitoring. Millstone 2 spent fuel pool silica measurements show typical values of 1.5 to 2.5 ppm silica over the last several years. These SFP silica concentrations are measured and monitored for any unusual trend. Further, Millstone has been a participant in the EPRI boraflex working group. As such we have used the EPRI RACKLIFE model to track the boraflex condition. RACKLIFE uses the measured silica concentrations, along with the individual boraflex panel irradiation history to determine individual boraflex panel degradation. The use of the EPRI RACKLIFE model independently confirms the conclusion that the Millstone 2 boraflex has undergone minimal thickness loss. RACKLIFE predicts the current peak gamma irradiation of the boraflex to be about 1.4 E+10 rads.

Based on the testing to date, DNC concludes that the boraflex contained in the Millstone 2 spent fuel racks has performed acceptably to date. There has been no detectable loss in boraflex thickness to date. While axial boraflex gaps are present due to irradiation caused shrinkage, the size of the gaps are small and have no appreciable reactivity impact to date. The proposed criticality analysis makes far more conservative assumptions on the boraflex condition than the existing criticality analysis of record, in case future in-service testing detects degradation.

DNC believes that the boraflex in the Millstone 2 racks has performed acceptably to date for 2 reasons: (1) because the boraflex material is 110 mills thick, which is thicker than typical, and (2) the boraflex material is well protected from interaction with water due to its design.

Region A Boraflex Poison Box Removal Analysis

It should be noted that the attached criticality analysis makes reference to analysis for removal of up to 2 boraflex poison boxes in each of the 3 storage racks that make up Region A. This analysis is for a future design change, and no approval is requested by the NRC for this condition.

Implementation Considerations

Proposed LCO 3.9.19 requires that prior to removal of a cell blocker, that the surrounding cells be vacated of fuel. A one-time exception is requested in the proposed TS LCO 3.9.19, to allow fuel to be adjacent to the removed cell blocker only during the initial installation of the Batch B fuel. This is justifiable in this circumstance because the only fuel which will be moved during the time of initial cell blocker removal are the Batch B fuel assemblies. These Batch B fuel assemblies can be stored anywhere in the spent fuel pool, thus a misloading event which causes a reactivity impact is not possible. This exception is requested to avoid unnecessary fuel movement and avoid the need to move a few fuel assemblies which have special handling requirements. Once the initial installation of Batch B fuel has occurred, this TS exception will no longer be applicable.

Safety Summary Conclusion

Implementation of the proposed changes are safe and will have essentially no effect on current plant operation. There are no hardware changes made to the plant due to these proposed changes. There are no changes in how fuel is moved, or the process used to qualify and verify fuel storage in the pool. The cell blockers are currently removable, therefore there are no modifications necessary to the cell blockers to allow storage of fuel under the cell blockers.

The spent fuel pool criticality analysis to support the proposed modifications uses standard criticality analysis methods. Millstone 2 has in the past normally maintained the spent fuel pool soluble boron concentration > 1720 ppm, so the imposition of a TS requirement to maintain this concentration is not new. A review of potential boron dilution events has shown that it is not credible that the spent fuel pool soluble boron concentration can be reduced from $\geq$ 1720 ppm to < 600 ppm.

The spent fuel racks were designed from a structural perspective for storage of fuel in all rack locations, and the current design basis heat load analysis already bounds the increased heat load from storing fuel in the 40 additional Region B locations.

The new fuel storage vault and fuel transfer machine have been previously qualified to a fuel enrichment of 5.0 w/o U-235, which bounds the current request.

The proposed criticality analysis for Millstone 2 credits boraflex for reactivity control. Millstone 2 has closely followed the condition of the fixed neutron absorber boraflex in the existing spent fuel racks. Measurements of the boraflex to date show that the material has performed acceptably. Additional and sufficient conservatism has been placed in the proposed criticality analysis to allow time for recovery should the in-service testing program show unexpected levels of degradation.

Attachment 2

Millstone Power Station, Unit No. 2

Technical Specifications Change Request 2-10-01
Fuel Pool Requirements
Significant Hazards Consideration

Technical Specification Change Request (TSCR) 2-10-01
Fuel Pool Requirements
Significant Hazards Consideration

Description of License Amendment Request

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to revise the Millstone Unit 2 Technical Specifications as described in this License Amendment Request. The proposed Technical Specification changes implement the following design changes:

- Increase the allowable nominal average fuel assembly enrichment from 4.5 w/o U-235 to 4.85 w/o U-235 for all regions of the spent fuel pool, the new fuel storage racks (dry), and the reactor core.
- Allow fuel to be located in 40 Region B storage cells which are currently empty and blocked. The cell blockers will be retained, and fuel is proposed to be stored under the cell blockers.
- Credit spent fuel pool soluble boron for reactivity control during normal conditions to maintain spent fuel pool $K_{eff} \leq 0.95$.

There are no physical changes in the plant hardware to implement these changes. Refer to Attachment 1 of this submittal for a detailed discussion of the proposed changes.

Basis for No Significant Hazards Consideration

In accordance with 10 CFR 50.92, DNC has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Previously evaluated Final Safety Analysis Report (FSAR) Chapter 14 accidents are a fuel handling accident either in the spent fuel pool or in containment, and a spent fuel cask drop accident.

   Since there are no changes to plant equipment, nor any changes in how fuel is moved, there are no changes to the probability of a fuel handling accident in the spent fuel pool or containment.

Since there are no changes to plant equipment, nor any changes in how a shielded cask would be moved, there are no changes to the probability of a spent fuel cask drop accident.

The consequences of a fuel drop accident in either containment or the spent fuel pool (SFP) are not affected, since none of the inputs to these fuel drop accidents is affected. There are no physical hardware changes made to the plant. The limiting fuel burnup is not changed, nor is there any change in the source term of radioactivity present in the fuel. Allowing fuel to be stored in the 40 Region B locations currently empty, does not alter the existing FSAR conclusion that a dropped fuel assembly or consolidated storage box could not strike more than one fuel assembly in the storage rack. This is still true since the fuel stored in these 40 locations is stored at the same elevation as fuel in any other storage locations. The FSAR states that the worst fuel handling incident that could occur in the SFP is the drop of a fuel assembly to the pool floor, with resultant failure of 14 fuel rods when the assembly rotates and impacts a protruding structure. Radiological consequences for both the failure of 14 rods and the entire fuel assembly are presented in the FSAR. The storage of fuel in the 40 currently blocked locations does not affect this FSAR sequence of events for the dropped fuel assembly in the SFP accident. The amount of soluble boron concentration necessary in the SFP to ensure that K_{eff} is maintained $\leq$ 0.95 on a 95/95 bases is increased from 800 ppm to 1400 ppm. However, this increase in required SFP soluble boron concentration does not increase any dose consequences from the fuel drop accident in the SFP. The increase in soluble boron concentration from 800 ppm to 1400 ppm is a result of crediting an additional 600 ppm of SFP soluble boron under normal conditions.

The consequences of a spent fuel cask drop accident in the spent fuel pool (SFP) is not affected, since none of the inputs to the spent fuel cask drop accident is affected. There are no physical hardware changes made to the plant. The limiting fuel burnup is not changed, nor is there any change in the source term of radioactivity present in the fuel. The amount of soluble boron concentration necessary in the SFP to ensure that K_{eff} is maintained $\leq$ 0.95 on a 95/95 bases is increased from 800 ppm to 1400 ppm. However, this increase in required SFP soluble boron concentration does not increase any dose consequences from the spent fuel cask drop accident in the SFP. The increase in soluble boron concentration from 800 ppm to 1400 ppm is a result of crediting an additional 600 ppm of SFP soluble boron under normal conditions.

With regard to the proposed change in the design features section of TS, which would allow higher enrichments in the new fuel storage (dry) vault, there are no FSAR Chapter 14 accident conditions currently analyzed, therefore there can be no change in probability or consequences of an existing accident.

With regard to the proposed change in the design features section of TS, which would allow higher enrichments in the reactor core, any significant increase in the probability or consequences of an accident previously analyzed would be evaluated by the specific reload design 10 CFR 50.59 evaluation. Should that reload design evaluation determine that there was a significant increase in the probability or consequences of an accident previously analyzed, that would then require NRC review.

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

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes to be made primarily affect nuclear criticality analysis and do not create a new or different kind of accident. Changes in allowed enrichment, boraflex credit, soluble boron credit, and allowing fuel to be stored in 40 additional locations are all impacts to the spent fuel pool criticality analysis. The SFP criticality analysis is part of the basic design of the system and is not an accident. The ability to maintain the spent fuel pool $K_{eff} \leq 0.95$, as well as within the 10 CFR 50 App. A GDC62 criteria of sub-critical have been evaluated. Criticality impacts are more appropriately discussed under the margin of safety criterion.

Since there are no changes to the plant equipment, there is no possibility of a new or different kind of accident being initiated or affected by equipment issues. There are no changes in how fuel is moved or qualified for storage, so a new accident can not be initiated from fuel handling related procedures.

Higher SFP soluble boron concentrations are required than previously required to compensate for the positive reactivity insertions from postulated accident conditions (i.e., dropped cask). However, merely increasing the amount of SFP soluble boron required for compensating for the existing analyzed accident does not create the potential for a new or different kind of accident.

Allowing SFP soluble boron to be credited under normal conditions to prevent criticality, as well as maintain SFP $K_{eff} \leq 0.95$, is new. Thus soluble boron is now credited in the prevention of criticality under normal conditions, just as fixed neutron poisons are credited. There is no FSAR Chapter 14 boron dilution event related to the spent fuel pool. A boron dilution event in the spent fuel pool could create dose consequences only if criticality were achieved. Prevention of a criticality accident is currently part of the basic design of the spent fuel racks under normal conditions and is not new. The criticality design of the spent fuel pool is such that should the SFP soluble boron concentration reach 0 ppm, the SFP will remain subcritical, including biases and uncertainties, therefore a criticality accident is not credible. Thus, a SFP boron dilution event does not have the potential to create a new or different kind of accident, since even if no SFP soluble boron remains, the SFP will still be sub-critical.

With regard to the proposed change in the design section of TS, which would allow higher enrichments in the new fuel storage (dry) vault, no new or different kind of accident conditions are created. The existing new fuel storage analysis previously submitted to the NRC is not altered, and already bounds enrichments up to 5.0 w/o U-235.

With regard to the proposed change in the design section of TS, which would allow higher enrichments in the reactor core, the creation of a new or different kind of accident condition would be evaluated by the specific reload design 10 CFR 50.59 evaluation. Should that reload design evaluation determine that there was the creation of a new or different kind of accident condition, that would then require NRC review and approval.

Therefore, based on the above analysis, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety relevant to the SFP are:

- to ensure that the SFP K_{eff} remains $\leq$ 0.95 on a 95/95 basis to ensure the criticality safety of the SFP.
- to ensure that the spent fuel in the SFP remains adequately cooled so that the fission product barriers remain intact.

A criticality analysis has been performed to ensure that the spent fuel pool K_{eff} remains $\leq$ 0.95 on a 95/95 basis under all normal and postulated accident conditions. Thus the margin of criticality safety is not changed. Most of the changes in the criticality analysis are of an input nature, such as a change in allowed enrichment. The only change in methodology is the crediting of soluble boron for normal conditions. The approach used is consistent with WCAP-14416-NP-A. The NRC has previously approved for other plants similar applications for soluble boron credit for normal conditions. The criticality analysis has been performed to ensure that the spent fuel pool K_{eff} remains less than 1.00 on a 95/95 basis even with 0 ppm soluble boron concentration in the SFP. This ensures compliance with GDC62.

The only change that could affect the SFP cooling analysis is allowing 40 additional fuel assemblies to be stored in the SFP. The current design basis heat load analysis already bounds the storage of these fuel assemblies. This ensures that the spent fuel in the SFP remains adequately cooled so that the fission product barriers remain intact. The current design basis heat load analysis bounds the increased fuel storage.

With regard to the proposed change in the design section of TS, which would allow higher enrichments in the new fuel storage (dry) vault, there is no significant reduction in the margin of safety. The existing new fuel storage analysis previously submitted and approved to the NRC is not altered, and already bounds enrichments up to 5.0 w/o U-235, to ensure that K_{eff} of the new fuel storage racks is maintained $\leq$ 0.95.

With regard to the proposed change in the design section of TS, which would allow higher enrichments in the reactor core, the verification that there is no significant reduction in the margin of safety would be evaluated by the specific reload design 10 CFR 50.59 evaluation. Should that reload design evaluation determine that there was a significant reduction in the margin safety, that would then require NRC review and approval.

Therefore, based on the above analysis, the proposed changes do not involve a reduction in a margin of safety.

Docket No. 50-336
B18501

Attachment 3

Millstone Power Station, Unit No. 2

Technical Specifications Change Request 2-10-01
Fuel Pool Requirements
Marked Up Pages

Technical Specifications Change Request 2-10-01
Marked Up Pages

Technical Specification Section Number	Title of Section	Page and Amendment Numbers	
LCO Index 3/4.9.17	Movement of Fuel in Spent Fuel Pool	IX	Amendment 249
LCO Index 3/4.9.18	Spent Fuel Pool - Reactivity Condition	IX	Amendment 249
Bases Index 3/4.9.17	Movement of Fuel in Spent Fuel Pool	XIV	Amendment 250
Bases Index 3/4.9.18	Spent Fuel Pool - Reactivity Condition	XIV	Amendment 250
3.9.16.2	Shielded Cask	3/4 9-20	Amendment 245
3.9.17	Movement of Fuel in Spent Fuel Pool	3/4 9-21	Amendment 245
3.9.18	SFP - Reactivity Condition	3/4 9-22	Amendment 172
Figure 3.9-1A	Min Req Fuel Assembly Exposure As a Function of Initial Enrichment To Permit Storage in Region C	3/4 9-23	Amendment 172
Figure 3.9-1B	Min Req Fuel Assembly Exposure As a Function of Initial Enrichment To Permit Storage in Region C w/Poison Pins Installed	3/4 9-23a	Amendment 172
Figure 3.9-2	SFP Arrangement	3/4 9-24	Amendment 172
Figure 3.9-4	Min Req Fuel Assembly Exposure As a Function of Initial Enrichment To Permit Storage in Region A	3/4 9-25a	Amendment 158
3.9.19	SFP - Storage Pattern	3/4 9-26	Amendment 172
5.3.1	Fuel Assemblies	5-4	Amendment 216
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5.6.3	Capacity	5-5a	Amendment 172
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Bases 3/4 9.17	Movement of Fuel in Spent Fuel Pool	B 3/4 9-3a	Amendment 245
Bases 3/4 9.18	SFP - Reactivity Condition	B 3/4 9-3a	Amendment 245
Bases 3/4 9.19	SFP - Storage Pattern	B 3/4 9-4	Amendment 172

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION

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3/4.9.11 and 3/4.9.12	WATER LEVEL - REACTOR VESSEL AND STORAGE POOL WATER LEVEL	B 3/4 9-2a
3/4.9.13	DELETED	B 3/4 9-3
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3/4.9.15	STORAGE POOL AREA VENTILATION SYSTEM	B 3/4 9-3
3/4.9.16	SHIELDED CASK	B 3/4 9-3a
3/4.9.17	~~MOVEMENT OF FUEL IN SPENT FUEL POOL~~ SPENT FUEL POOL BORON CONCENTRATION	B 3/4 9-3 a
3/4.9.18	SPENT FUEL POOL - ~~REACTIVITY CONDITION~~ STORAGE	B 3/4 9-3 b
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SHIELDED CASK

LIMITING CONDITION FOR OPERATION

3.9.16.2 Prior to movement of a shielded cask over the spent fuel pool cask laydown area, the boron concentration of the pool shall be greater than or equal to 800 parts per million (ppm).

APPLICABILITY: Whenever a shielded cask is to be moved over the spent fuel pool cask laydown area.

ACTION:

With the boron concentration less than 800 ppm, suspend all movement of the shielded cask over the spent fuel pool cask laydown area.

SURVEILLANCE REQUIREMENTS

4.9.16.2 Verify that the boron concentration is greater than or equal to 800 ppm within 24 hours prior to any movement of a shielded cask over the spent fuel pool cask laydown area.

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April 28, 2000

~~MOVEMENT OF FUEL IN~~ SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.17 Prior to movement of a fuel assembly, or a consolidated fuel storage box, in the spent fuel pool, the boron concentration of the pool shall be greater than or equal to 800 ppm.

APPLICABILITY: Whenever a fuel assembly, or a consolidated fuel storage box, is moved in the spent fuel pool.

ACTION:

With the boron concentration less than 800 ppm, suspend the movement of all fuel in the spent fuel pool.

SURVEILLANCE REQUIREMENTS

4.9.17 Verify that the boron concentration is greater than or equal to 800 ppm within 24 hours prior to any movement of a fuel assembly, or a consolidated fuel storage box, in the spent fuel pool and every 72 hours thereafter.

Replace with INSERT A

INSERT A

To Page 3/4 9-21

3.9.17	The boron concentration in the spent fuel pool shall be greater than or equal to 1720 parts per million (ppm).
APPLICABILITY	Whenever any fuel assembly or consolidated fuel storage box is stored in the spent fuel pool.
ACTION:	With the boron concentration less than 1720 ppm, suspend the movement of all fuel, consolidated fuel storage boxes, and shielded casks, and immediately initiate action to restore the spent fuel pool boron concentration to within its limit. The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.17	Verify that the boron concentration is greater than or equal to 1720 ppm every 7 days, and within 24 hours prior to the initial movement of a fuel assembly or consolidated fuel storage box in the Spent Fuel Pool, or shielded cask over the cask laydown area.

SPENT FUEL POOL-~~REACTIVITY CONDITION~~ STORAGE

Replace with Insert B

LIMITING CONDITION FOR OPERATION

3.9.18 The Reactivity Condition of the spent fuel pool shall be such that K_{eff} is less-than-or-equal-to 0.95 at all times.

APPLICABILITY: Whenever fuel is in the spent fuel pool.

ACTION:

Borate until $K_{eff} \leq .95$ is reached.

SURVEILLANCE REQUIREMENTS

4.9.18.1 Ensure that all fuel assemblies to be placed in Region C (as shown in Figure 3.9-2) of the spent fuel pool satisfy either:

(a) Fuel assembly enrichment and burnup are within the limits of Figure 3.9-1a by checking the assembly's design and burnup documentation; or

(b) Fuel assembly enrichment and burnup are within the limits of Figure 3.9-1b by checking the assembly's design and burnup documentation, and borated stainless steel poison pins are installed in the assembly's center guide tube and in two diagonally opposite guide tubes.

4.9.18.2 Ensure that the contents of each consolidated fuel storage box to be placed in Region C (as shown in Figure 3.9-2) of the spent fuel pool are within the enrichment and burn-up limits of Figure 3.9-3 by checking the design and burn-up documentation for storage box contents.

4.9.18.3 Ensure that all fuel assemblies to be placed in Region A (as shown in Figure 3.9-2) of the spent fuel pool are within the enrichment and burnup limits of Figure 3.9-4 by checking the assembly's design and burnup documentation.

0744

INSERT B

To Page 3/4 9-22

3.9.18 The following spent fuel pool storage requirement will be met:

(a) The combination of initial enrichment and burnup of each fuel assembly stored in Region A shall be within the acceptable burnup domain of Figure 3.9-4; and

(b) (1) The combination of initial enrichment and burnup of a fuel assembly stored in Region C shall be within the acceptable burnup domain of Figure 3.9-1A;

OR

(2) The combination of initial enrichment and burnup of a fuel assembly stored in Region C shall be within the acceptable burnup domain of Figure 3.9-1B, and borated stainless steel poison pins are installed in the assembly's center guide tube and in two diagonally opposite guide tubes; and

(c) The combination of initial enrichment and burnup of each consolidated fuel storage box stored in Region C shall be within the acceptable burnup domain of Figure 3.9-3.

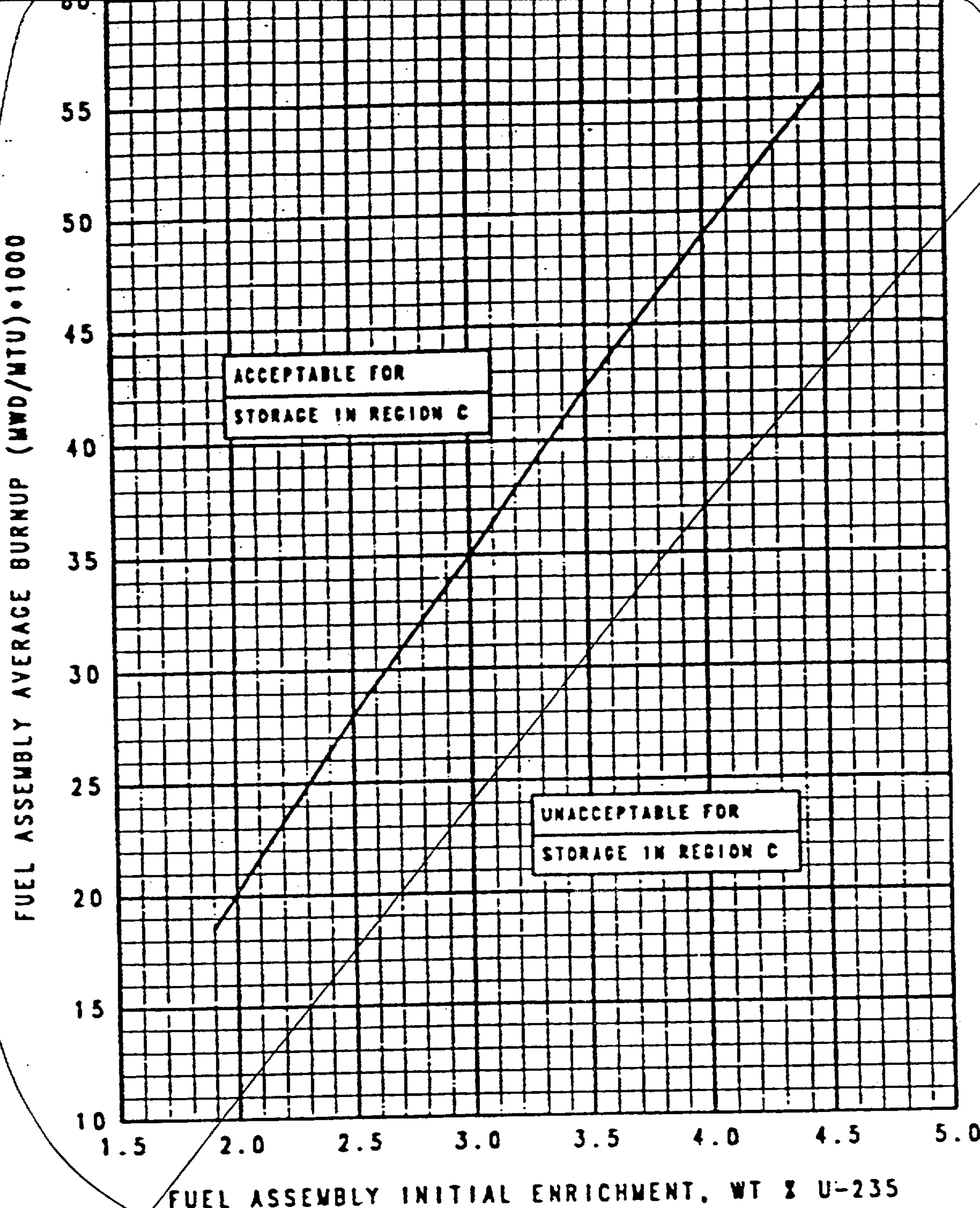
APPLICABILITY Whenever any fuel assembly or consolidated fuel storage box is stored in the spent fuel pool.

ACTION: Immediately initiate action to move the non-complying fuel assembly or consolidated fuel storage box to an acceptable location.

The provisions of specification 3.0.3 are not applicable.

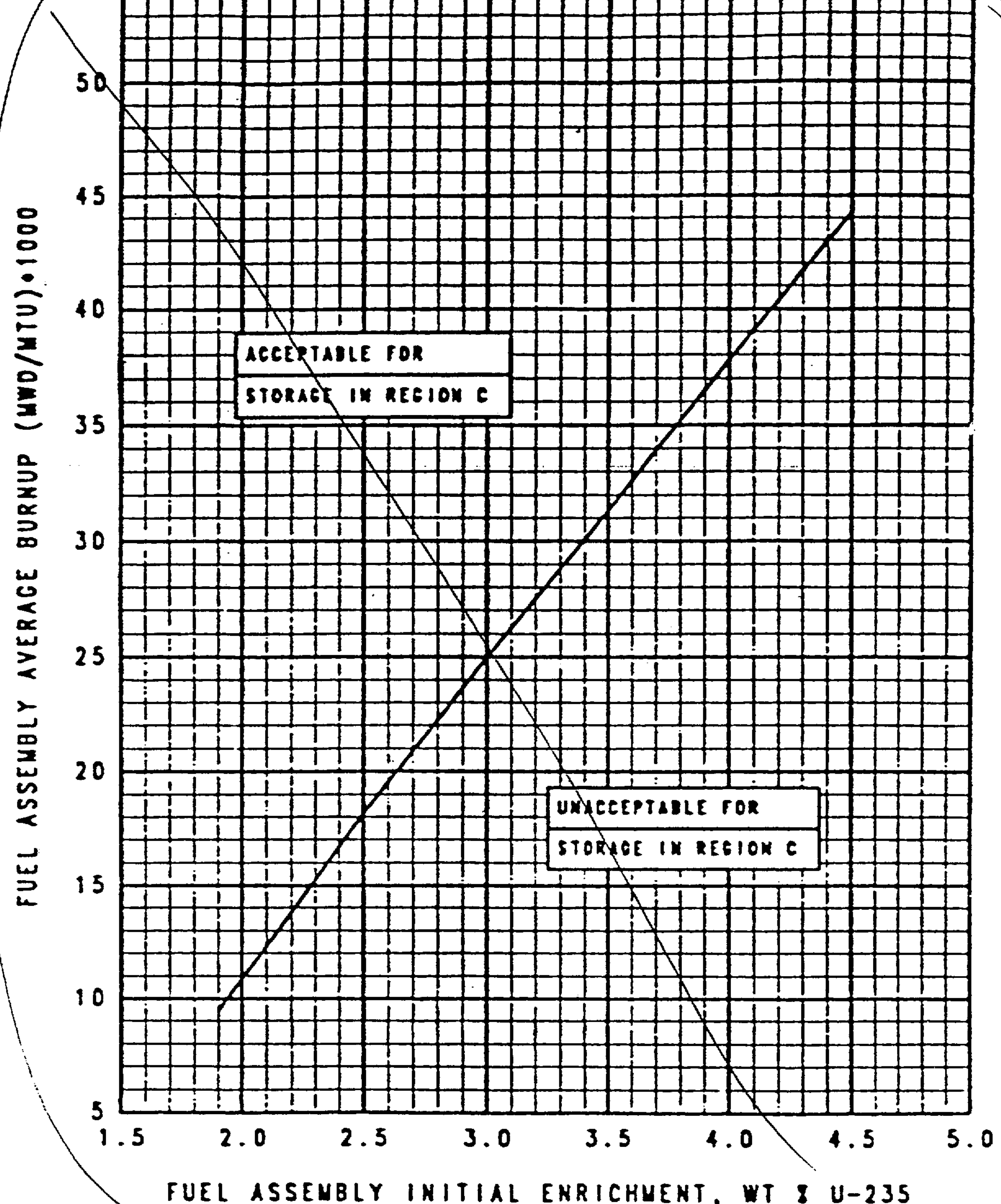
SURVEILLANCE REQUIREMENTS

4.9.18 Prior to storing a fuel assembly or consolidated fuel storage box in the spent fuel racks, verify by administrative means the initial enrichment and burnup of the fuel assembly or consolidated fuel storage box is in accordance with the acceptable specifications for that Storage Region.



(Replaced with ~~[illegible]~~ new Figure 3.9-1A

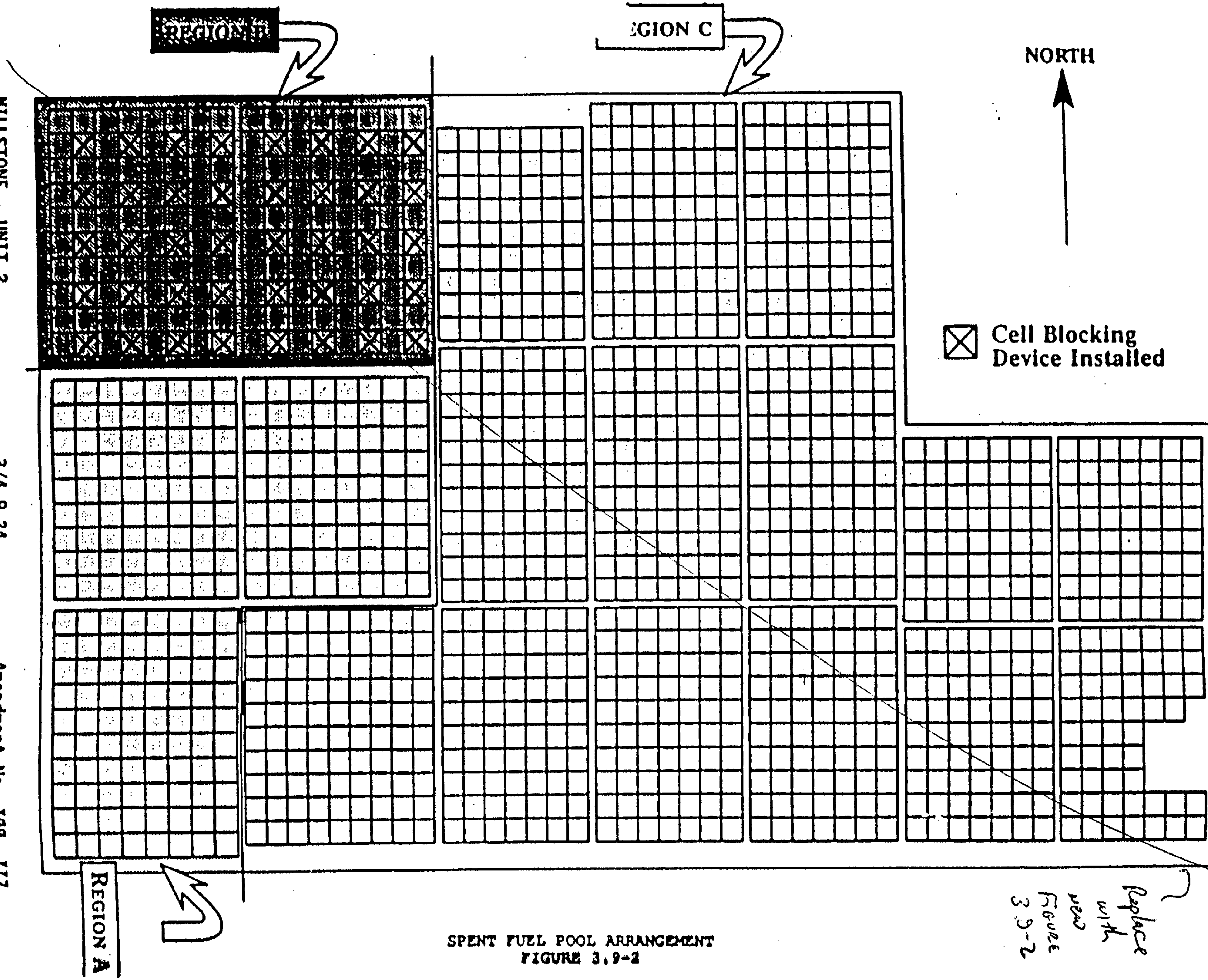
FIGURE 3.9-1A MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION C



Replace with new FIGURE 3.9-1B

FIGURE 3.9-1B MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION C WITH POISON PINS INSTALLED

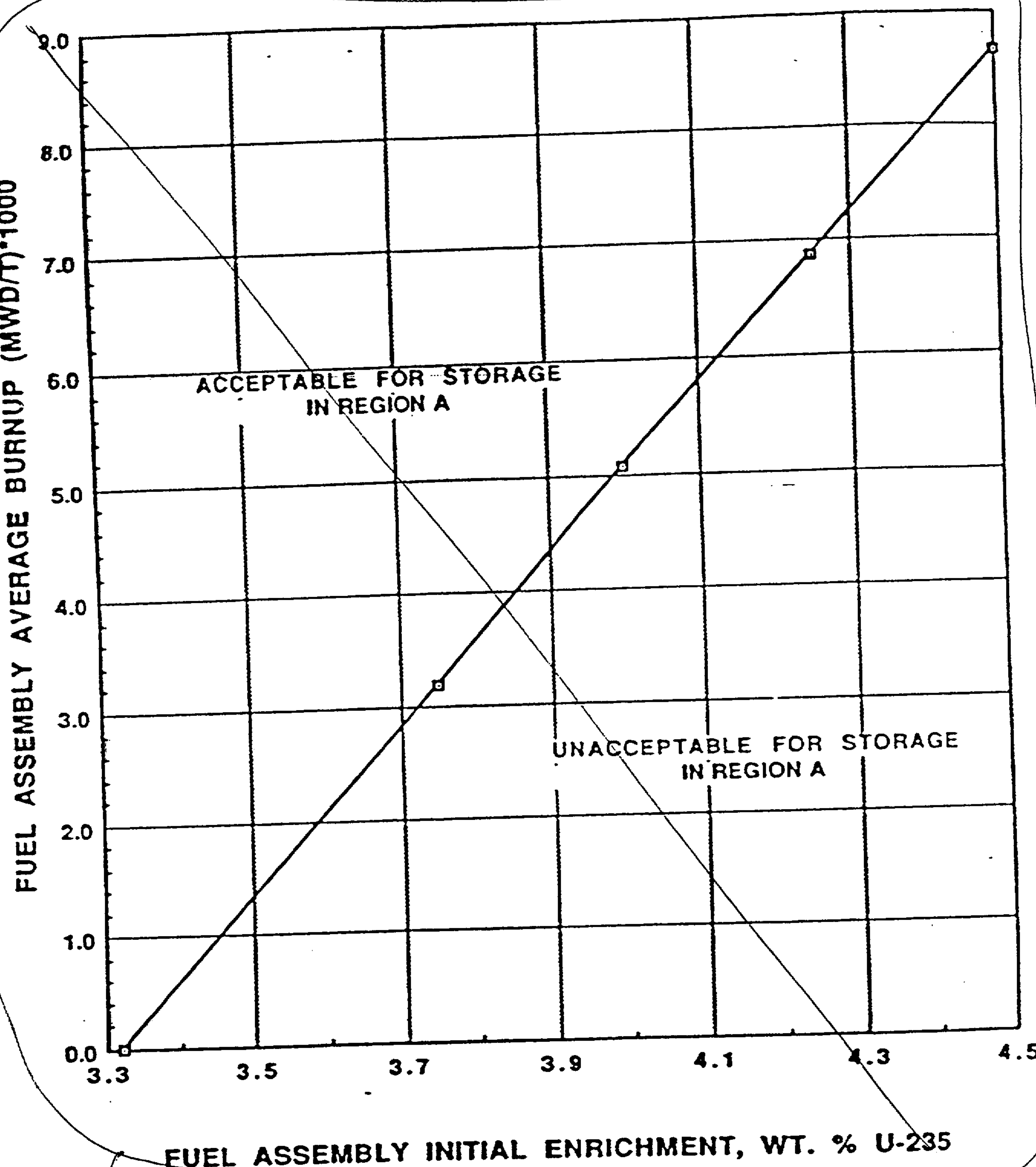
MILLSTONE - UNIT 2



SPENT FUEL POOL ARRANGEMENT
FIGURE 3.9-2

3/4 9-24

Amendment No. ~~109~~, ~~117~~, 155, 172.



Replace with new Figure 3.9-4

FIG. 3.9-4 MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION A

SPENT FUEL POOL - STORAGE PATTERN

LIMITING CONDITION FOR OPERATION

3.9.19 Each STORAGE PATTERN of the Region B spent fuel pool racks shall require that:

(1) A cell blocking device is installed in those cell locations shown in Figure 3.9-2; or

(2) If a cell blocking device has been removed, all cells in the STORAGE PATTERN, must be vacant of stored fuel assemblies, except the location with the removed cell blocking device,

APPLICABILITY: Fuel in the spent fuel pool.**

ACTION:

Take immediate action to comply with either 3.9.19(1) or (2).
The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.19 Verify that 3.9.19 is satisfied prior to removing a cell blocking device.

. The blocked location may store a Batch B fuel assembly * underneath the cell blocker.

* A Batch B fuel assembly refers to any of the Batch B fuel assemblies which were part of the first Millstone 2 core.

** This LCO is not applicable during the initial installation of Batch B fuel assemblies in the cell blocker locations.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 54 psig and an equilibrium liner temperature of 289°F.

PENETRATIONS

5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the design provisions contained in Section 5.2.8 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing 176 rods. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of ~~4.5~~ weight percent of U-235.

nominal average 4.85

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and no part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the design provisions contained in Section 3.0 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F except for the pressurizer which is 700°F.

A fuel rod shall have a maximum enrichment of 5.0 weight percent of U-235.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is a nominal 10,981 ft^3.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

replace with Insert #C

CRITICALITY

~~5.6.1 a) The new fuel (dry) storage racks are designed and shall be maintained with sufficient center to center distance between assemblies to ensure a $k_{eff} \leq .95$. The maximum nominal fuel enrichment to be stored in these racks is 4.50 weight percent of U-235.~~

~~b) Region A of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations to ensure a $K_{eff} \leq .95$ with the storage pool filled with unborated water. Fuel assemblies stored in this region must comply with Figure 3.9-4 to ensure that the design burnup has been sustained.~~

~~c) Region B of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center-to-center distance between storage locations to ensure $K_{eff} \leq .95$ with a storage pool filled with unborated water. Fuel assemblies stored in this region may have a maximum nominal enrichment of 4.5 weight percent U-235. Fuel assemblies stored in this region are placed in a 3 out of 4 STORAGE PATTERN for reactivity control.~~

~~d) Region C of the spent fuel storage pool is designed and shall be maintained with a 9.0 inch center to center distance between storage locations to ensure a $K_{eff} \leq .95$ with the storage pool filled with unborated water. Fuel assemblies stored in this region must comply with Figures 3.9-1a or 3.9-1b to ensure that the design burn-up has been sustained. Additionally, fuel assemblies utilizing Figure 3.9-1b require that borated stainless steel poison pins are installed in the fuel assembly's center guide tube and in two diagonally opposite guide tubes. The poison pins are solid 0.87 inch O.D. borated stainless steel, with a boron content of 2 weight percent boron.~~

~~e) Region C of the spent fuel storage pool is designed to permit storage of consolidated fuel and ensure a $K_{eff} \leq 0.95$. The contents of consolidated fuel storage boxes to be stored in this region must comply with Figure 3.9-3.~~

Unit 2
Technical Specification Changes

Insert C

Change 5.6 FUEL STORAGE Page 5-5 to read…

CRITICALITY

5.6.1 a) The new fuel (dry) storage racks are designed and shall be maintained with sufficient center to center distance between assemblies to ensure a $k_{eff} \leq .95$. The maximum nominal average fuel assembly enrichment to be stored in these racks is 4.85 weight percent ~~U_{235}~~. The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent ~~U_{235}~~.

U-235

b) The spent fuel storage racks are designed and shall be maintained with fuel assemblies having a maximum nominal average enrichment of 4.85 weight percent ~~U_{235}~~. The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent ~~U_{235}~~.

c) The spent fuel storage racks are designed and shall be maintained with $k_{eff} \leq 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties.

d) The spent fuel storage racks are designed and shall be maintained with $k_{eff} \leq .95$ if fully flooded with water borated to 600 ppm, which includes an allowance for uncertainties.

e) Region A of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figure 3.9-4 to ensure that the design burnup has been sustained.

f) Region B of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Region B contains both blocked and un-blocked storage locations, shown in Figure 3.9-2. Fuel having a maximum nominal enrichment of 4.85 weight percent ~~U_{235}~~, with no burnup may be stored in un-blocked locations. Fuel stored in blocked locations must be Batch B fuel assemblies.

U-235

g) Region C of the spent fuel storage pool is designed and shall be maintained with a 9.0 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figures 3.9-1a or 3.9-1b to ensure that the design burn-up has been sustained. Additionally, fuel assemblies utilizing Figure 3.9-1b require that borated stainless steel poison pins are installed in the fuel assembly's center guide tube and in two diagonally opposite guide tubes. The poison pins are solid 0.87 inch O.D. borated stainless steel, with a boron content of 2 weight percent boron.

h) Region C of the spent fuel storage pool is designed to permit storage of consolidated fuel. The contents of the consolidated fuel storage boxes to be stored in this region must comply with Figure 3.9-3 to ensure that the design burnup has been sustained.

DESIGN FEATURES

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 22'6".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 224 storage locations in Region A, 160 storage locations in Region B and 962 storage locations in Region C for a total of 1346 storage locations~~*~~.

~~*This translates into 1306 storage locations to receive spent fuel and 40 storage locations to remain blocked.~~

The spent fuel pool area access doors and other openings, required to be closed, are listed in the Technical Requirements Manual.

The Millstone Unit No. 2 Auxiliary Building elevator shaft smoke/heat hole has been evaluated and determined to be an acceptable minor leakage pathway. Therefore, spent fuel pool area integrity is maintained, and the required Enclosure Building Filtration Train is OPERABLE, when the elevator shaft smoke/heat hole is open. 2-HV-171, Spent Fuel Pool Area Exhaust Damper, is not an acceptable bypass leakage path and must remain closed when necessary to maintain spent fuel pool area integrity.

The laboratory testing requirement for the charcoal sample to have a removal efficiency of $\geq$ 95% is more conservative than the elemental and organic iodine removal efficiencies of 90% and 70%, respectively, assumed in the DBA analyses for the EBFS charcoal adsorbers in the Millstone Unit 2 Final Safety Analysis Report. A removal efficiency acceptance criteria of $\geq$ 95% will ensure the charcoal has the capability to perform its intended safety function throughout the length of an operating cycle.

3/4.9.16 SHIELDED CASK

The limitations of this specification and 3/4.9.15 ensure that in the event of a shielded cask drop accident ~~1)~~ the doses from ruptured fuel assemblies will be within the assumptions of the safety analyses~~, 2) K_{eff} will remain ≤ .95~~.

~~3/4.9.17 MOVEMENT OF FUEL IN SPENT FUEL POOL~~

~~The limitations of this specification ensure that in the event of a fuel handling accident involving a dropped, misplaced, or misloaded fuel assembly (or consolidated fuel storage box), the Keff of the spent fuel pool racks and fuel transfer carriage will remain less than or equal to 0.95.~~

INSERT D

3/4.9.18 SPENT FUEL POOL - ~~REACTIVITY CONDITION~~ STORAGE

The limitations described by Figures 3.9-1a, 3.9-1b, and 3.9-3 ensure that the reactivity of fuel assemblies and consolidated fuel storage boxes, introduced into the Region C spent fuel racks, are conservatively within the assumptions of the safety analysis.

The limitations described by Figure 3.9-4 ensure that the reactivity of the fuel assemblies, introduced into the Region A spent fuel racks, are conservatively within the assumptions of the safety analysis.

Unit 2
Technical Specification Changes

Insert D

Change 3 / 4.9.17 MOVEMENT OF FUEL IN SPENT FUEL POOL page B 3 / 4 9-3a to...

3 / 4.9.17 SPENT FUEL POOL BORON CONCENTRATION

The limitations of this specification ensures that sufficient boron is present to maintain spent fuel pool $k_{eff} \leq 0.95$ under accident conditions.

Postulated accident conditions which could cause an increase in spent fuel pool reactivity are: a single dropped or mis-loaded fuel assembly, a single dropped or mis-loaded Consolidated Fuel Storage Box, or a shielded cask drop onto the storage racks. A spent fuel pool soluble boron concentration of 1400 ppm is sufficient to ensure $k_{eff} \leq 0.95$ under these postulated accident conditions. The required spent fuel pool soluble boron concentration of $\geq$ 1720 ppm conservatively bounds the required 1400 ppm. The ACTION statement ensures that if the soluble boron concentration falls below the required amount, that fuel movement or shielded cask movement is stopped, until the boron concentration is restored to within limits.

An additional basis of this LCO is to establish 1720 ppm as the minimum spent fuel pool soluble boron concentration which is sufficient to ensure that the design basis value of 600 ppm soluble boron is not reached due to a postulated spent fuel pool boron dilution event. As part of the spent fuel pool criticality design, a spent fuel pool soluble boron concentration of 600 ppm is sufficient to ensure $k_{eff} \leq 0.95$, provided all fuel is stored consistent with LCO requirements. By maintaining the spent fuel pool soluble boron concentration $\geq$ 1720 ppm, sufficient time is provided to allow the operators to detect a boron dilution event, and terminate the event, prior to the spent fuel pool being diluted below 600 ppm. In the unlikely event that the spent fuel pool soluble boron concentration is decreased to 0 ppm, k_{eff} will be maintained < 1.00, provided all fuel is stored consistent with LCO requirements. The ACTION statement ensures that if the soluble boron concentration falls below the required amount, that immediate action is taken to restore the soluble boron concentration to within limits, and that fuel movement or shielded cask movement is stopped. Fuel movement and shielded cask movement is stopped to prevent the possibility of creating an accident condition at the same time that the minimum soluble boron is below limits for a potential boron dilution event.

The surveillance of the spent fuel pool boron concentration within 24 hours of fuel movement, consolidated fuel movement, or cask movement over the cask laydown area, verifies that the boron concentration is within limits just prior to the movement. The 7 day surveillance interval frequency is sufficient since no deliberate major replenishment of pool water is expected to take place over this short period of time.

3/4.9.19 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity condition of the Region B storage racks and spent fuel pool K_{eff} will remain less than or equal to 0.95. ; or a Batch B fuel assembly may be stored in the blocked location,

The Cell Blocking Devices in the 4th location of the Region B storage racks are designed to prevent inadvertent placement and/or storage in the blocked locations. The blocked location remains empty ~~to provide the flux trap~~ to maintain reactivity control for fuel assembly storage in any adjacent locations. Region B is designed for the storage of new assemblies in the spent fuel pool, and for fuel assemblies which have not sustained sufficient burnup to be stored in Region A or Region C. (non-cell blocker locations)

Add Insert E

3/4.9.20 SPENT FUEL POOL - CONSOLIDATION

The limitations of these specifications ensure that the decay heat rates and radioactive inventory of the candidate fuel assemblies for consolidation are conservatively within the assumptions of the safety analysis.

Unit 2
Technical Specification Changes

Insert E

Batch B

This LCO is not applicable during the initial installation of Batch B fuel assemblies in the cell blocker locations of Region B. This is acceptable because only Region B fuel assemblies will be moved during the initial installation of Batch B fuel assemblies under the Region B cell blockers. Batch B fuel assemblies are qualified for storage in any spent fuel pool storage rack location, hence a fuel misloading event which causes a reactivity consequence is not credible. This exception is valid only during the initial installation of Batch B fuel assemblies in the cell blocker locations.

Docket No. 50-336
B18501

Attachment 4

Millstone Power Station, Unit No. 2

Technical Specifications Change Request 2-10-01
Fuel Pool Requirements
Retyped Pages

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION

PAGE

3/4.9 REFUELING OPERATIONS

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

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SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.17 The boron concentration in the spent fuel pool shall be greater than or equal to 1720 parts per million (ppm).

APPLICABILITY: Whenever any fuel assembly or consolidated fuel storage box is stored in the spent fuel pool.

ACTION:

With the boron concentration less than 1720 ppm, suspend the movement of all fuel, consolidated fuel storage boxes, and shielded casks, and immediately initiate action to restore the spent fuel pool boron concentration to within its limit.

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.17 Verify that the boron concentration is greater than or equal to 1720 ppm every 7 days, and within 24 hours prior to the initial movement of a fuel assembly or consolidated fuel storage box in the Spent Fuel Pool, or shielded cask over the cask laydown area.

SPENT FUEL POOL-STORAGE

LIMITING CONDITION FOR OPERATION

3.9.18 The following spent fuel pool storage requirement will be met:

(a) The combination of initial enrichment and burnup of each fuel assembly stored in Region A shall be within the acceptable burnup domain of Figure 3.9-4; and

(b)(1) The combination of initial enrichment and burnup of a fuel assembly stored in Region C shall be within the acceptable burnup domain of Figure 3.9-1A;

OR

(2) The combination of initial enrichment and burnup of a fuel assembly stored in Region C shall be within the acceptable burnup domain of Figure 3.9-1B, and borated stainless steel poison pins are installed in the assembly's center guide tube and in two diagonally opposite guide tubes; and

(c) The combination of initial enrichment and burnup of each consolidated fuel storage box stored in Region C shall be within the acceptable burnup domain of Figure 3.9-3.

APPLICABILITY: Whenever any fuel assembly or consolidated fuel storage box is stored in the spent fuel pool.

ACTION:

Immediately initiate action to move the non-complying fuel assembly or consolidated fuel storage box to an acceptable location.

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.18 Prior to storing a fuel assembly or consolidated fuel storage box in the spent fuel racks, verify by administrative means the initial enrichment and burnup of the fuel assembly or consolidated fuel storage box is in accordance with the acceptable specifications for that Storage Region.

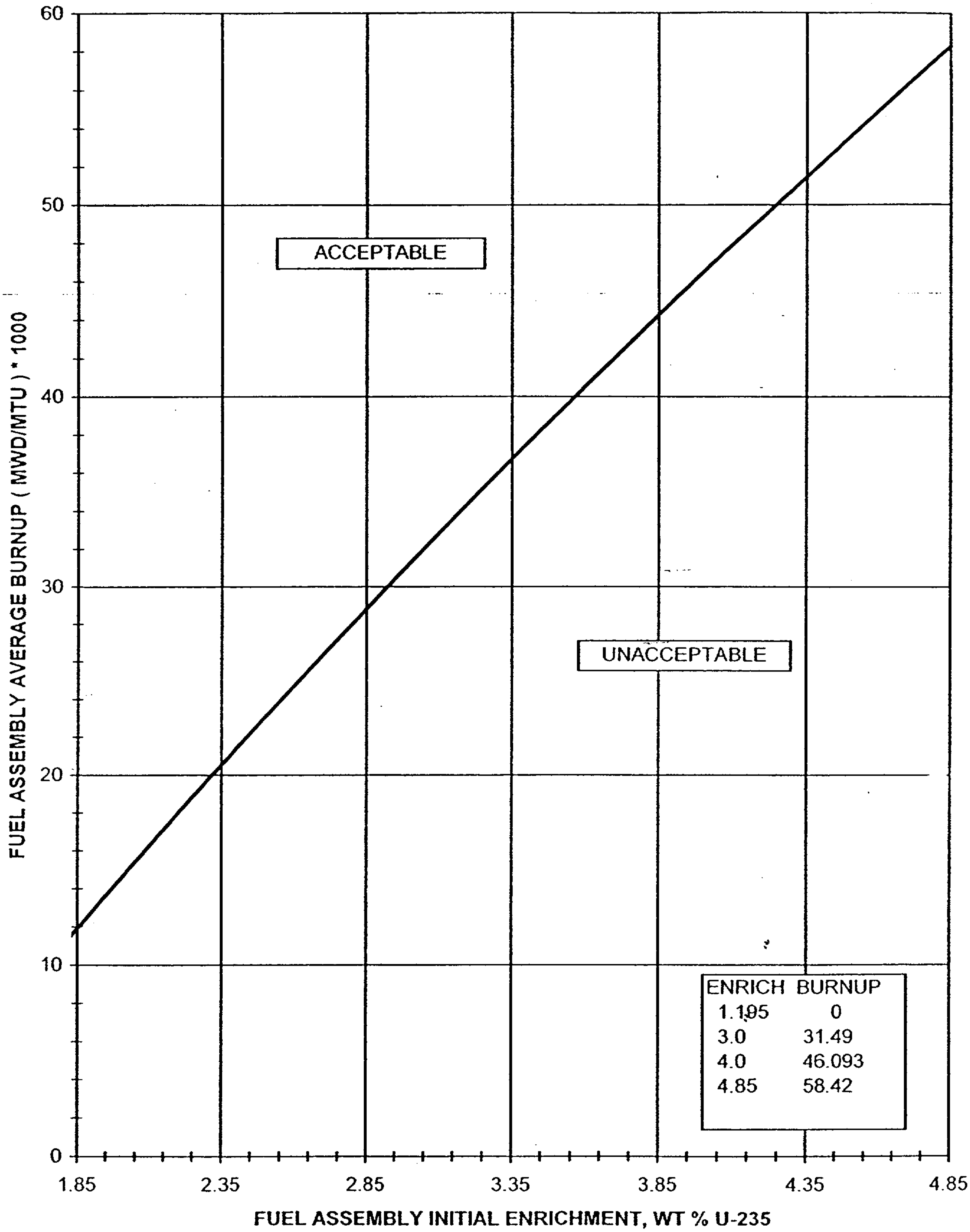


FIGURE 3.9-1A MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION C

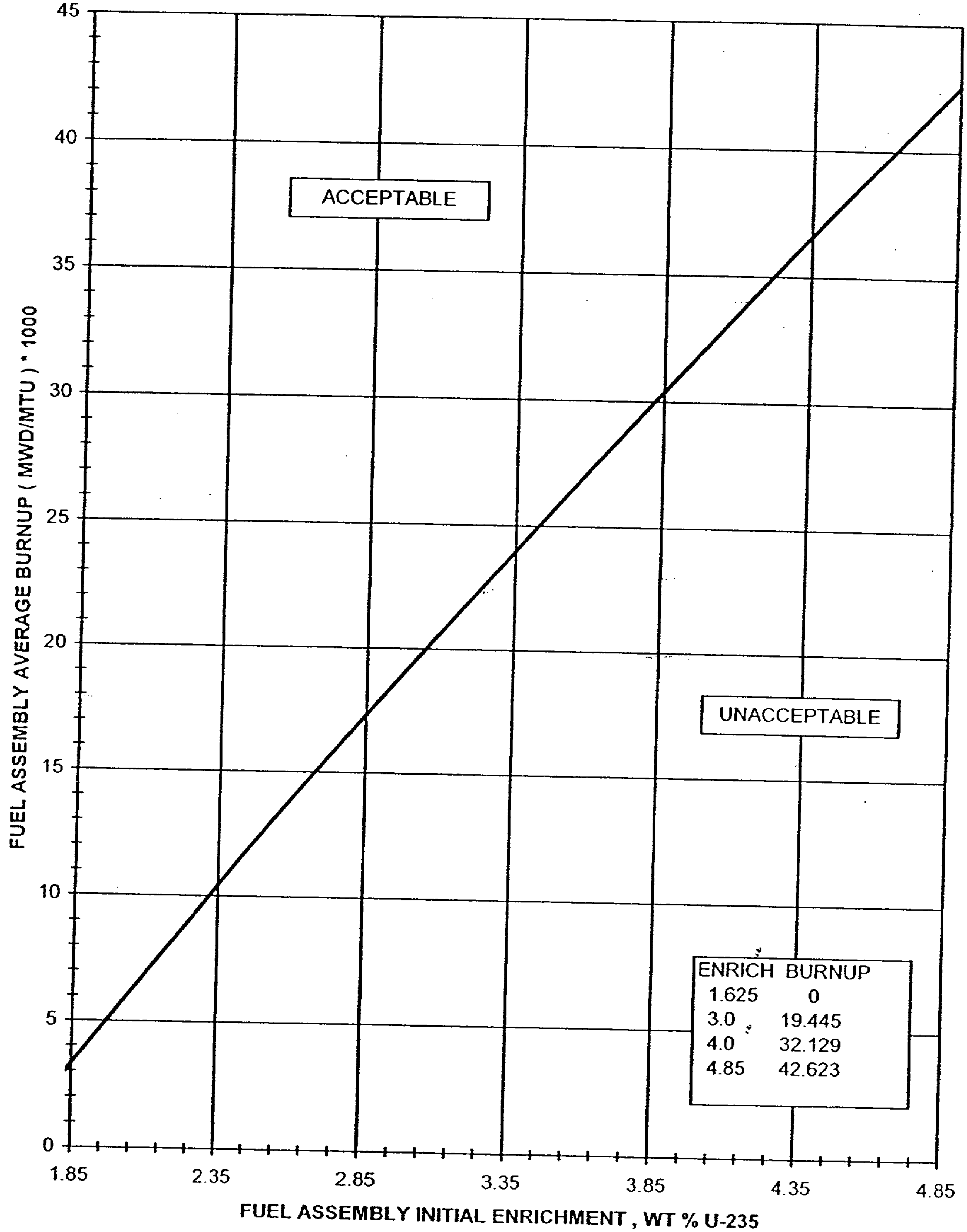
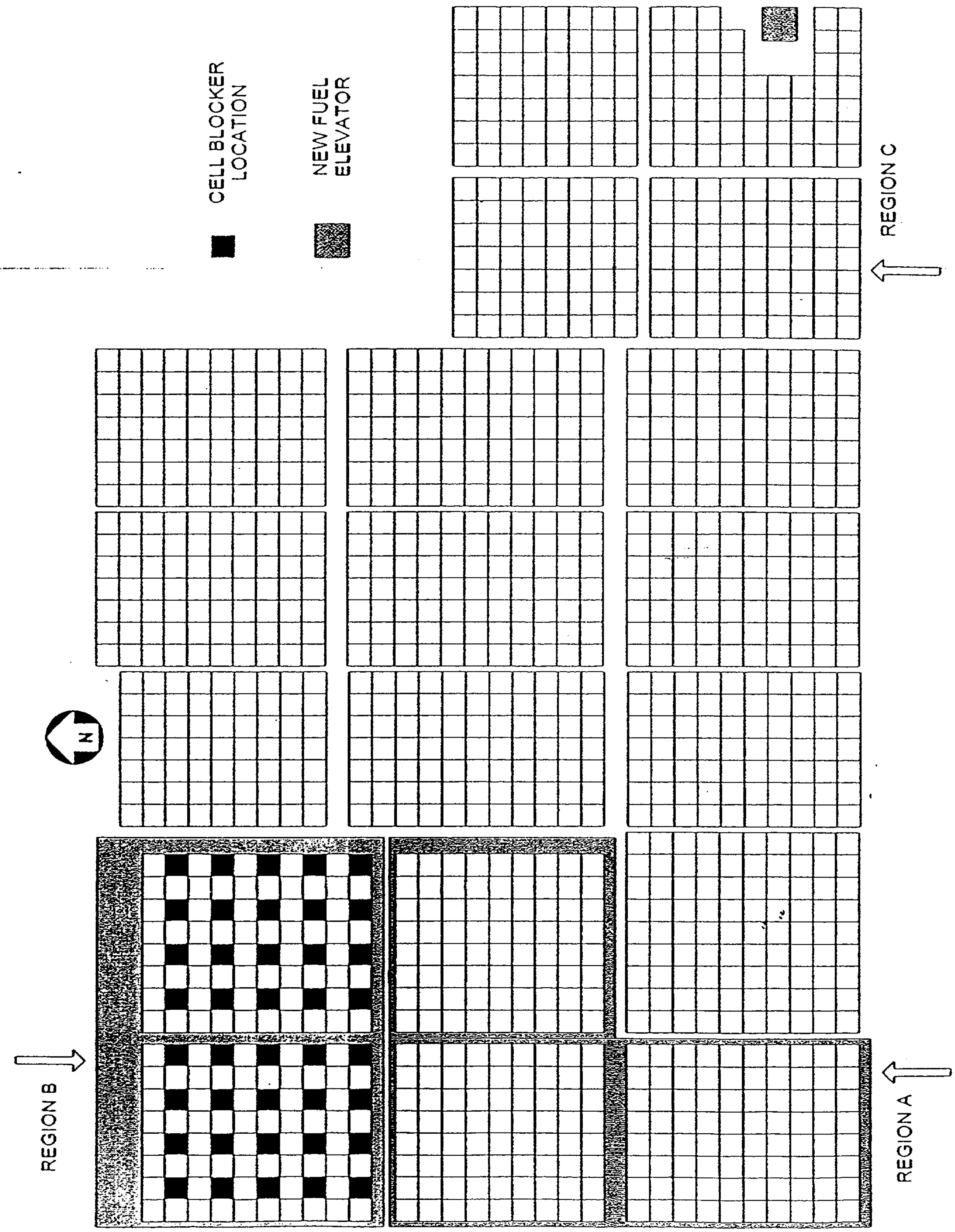


FIGURE 3.9-1B MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION C WITH POISON PINS INSTALLED

SPENT FUEL POOL ARRANGEMENT
FIGURE 3.9-2
(NOT TO SCALE)



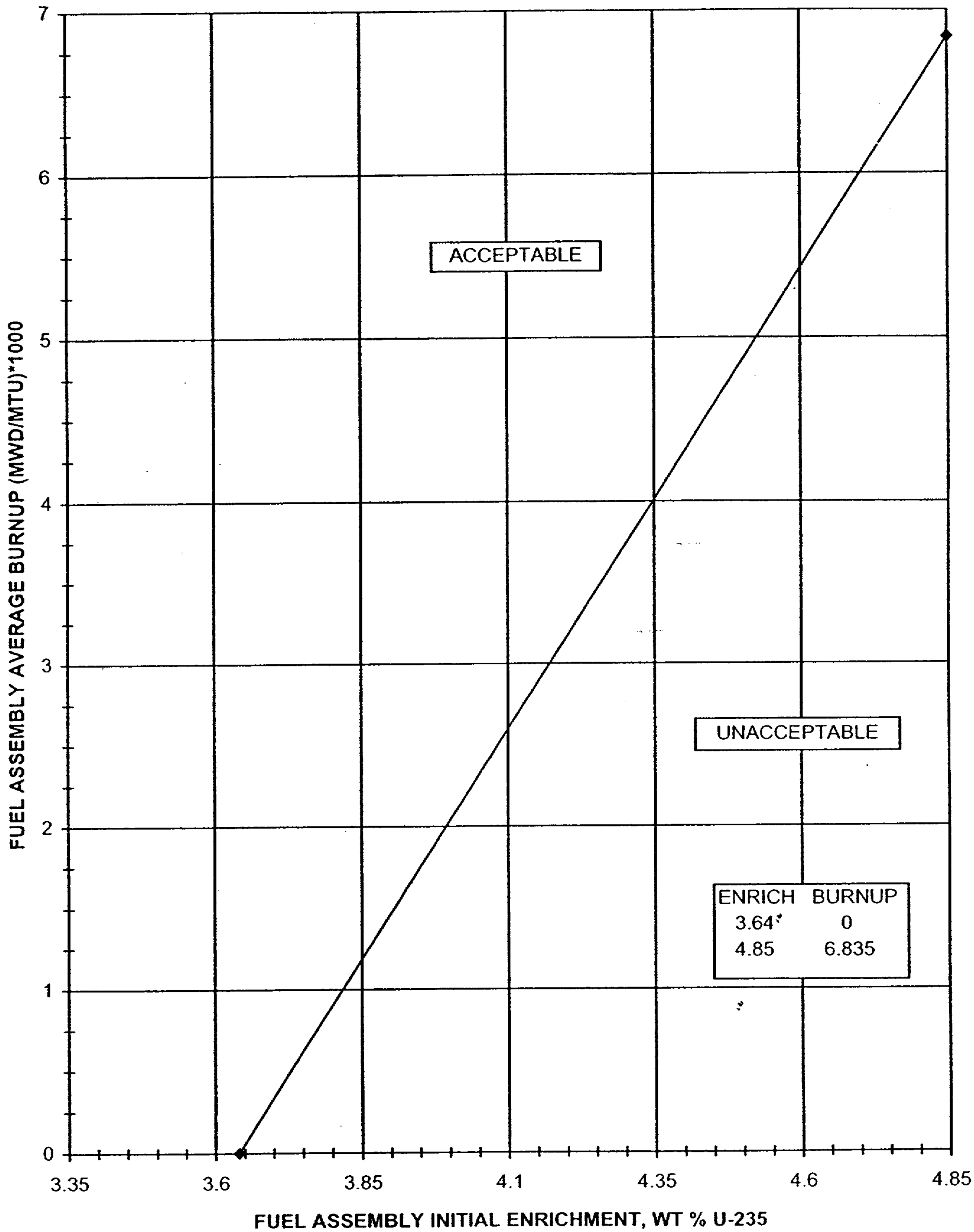


FIGURE 3.9-4 MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION A

SPENT FUEL POOL - STORAGE PATTERN

LIMITING CONDITION FOR OPERATION

3.9.19 Each STORAGE PATTERN of the Region B spent fuel pool racks shall require that:

(1) A cell blocking device is installed in those cell locations shown in Figure 3.9-2. The blocked location may store a Batch B fuel assembly* underneath the cell blocker; or

(2) If a cell blocking device has been removed, all cells in the STORAGE PATTERN, except the location with the removed cell blocking device, must be vacant of stored fuel assemblies.

APPLICABILITY: Fuel in the spent fuel pool.**

ACTION:

Take immediate action to comply with either 3.9.19(1) or (2).

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.19 Verify that 3.9.19 is satisfied prior to removing a cell blocking device.

*A Batch B fuel assembly refers to any of the Batch B fuel assemblies which were part of the first Millstone 2 core.

**This LCO is not applicable during the initial installation of Batch B fuel assemblies in the cell blocker locations.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 54 psig and an equilibrium liner temperature of 289°F.

PENETRATIONS

5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the design provisions contained in Section 5.2.8 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing 176 rods. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum nominal average enrichment of 4.85 weight percent of U-235. A fuel rod shall have a maximum enrichment of 5.0 weight percent of U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and no part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the design provisions contained in Section 3.0 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is a nominal 10,981 ft^3.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a) The new fuel (dry) storage racks are designed and shall be maintained with sufficient center to center distance between assemblies to ensure a $k_{eff} \leq .95$. The maximum nominal average fuel assembly enrichment to be stored in these racks is 4.85 weight percent U-235. The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent U-235.

b) The spent fuel storage racks are designed and shall be maintained with fuel assemblies having a maximum nominal average enrichment of 4.85 weight percent U-235. The maximum fuel rod enrichment to be stored in these racks is 5.0 weight percent U-235.

c) The spent fuel storage racks are designed and shall be maintained with $K_{eff} \leq 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties.

d) The spent fuel storage racks are designed and shall be maintained with $K_{eff} \leq .95$ if fully flooded with water borated to 600 ppm, which includes an allowance for uncertainties.

e) Region A of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figure 3.9-4 to ensure that the design burnup has been sustained.

f) Region B of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Region B contains both blocked and un-blocked storage locations, shown in Figure 3.9-2. Fuel having a maximum nominal enrichment of 4.85 weight percent U-235, with no burnup may be stored in un-blocked locations. Fuel stored in blocked locations must be Batch B fuel assemblies.

g) Region C of the spent fuel storage pool is designed and shall be maintained with a 9.0 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figures 3.9-1a or 3.9-1b to ensure that the design burn-up has been sustained. Additionally, fuel assemblies utilizing Figure 3.9-1b require that borated stainless steel poison pins are installed in the fuel assembly's center guide tube and in two diagonally opposite guide tubes. The poison pins are solid 0.87 inch O.D. borated stainless steel, with a boron content of 2 weight percent boron.

h) Region C of the spent fuel storage pool is designed to permit storage of consolidated fuel. The contents of the consolidated fuel storage boxes to be stored in this region must comply with Figure 3.9-3 to ensure that the design burnup has been sustained.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 22′6".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 224 storage locations in Region A, 160 storage locations in Region B and 962 storage locations in Region C for a total of 1346 storage locations.

The spent fuel pool area access doors and other openings, required to be closed, are listed in the Technical Requirements Manual.

The Millstone Unit No. 2 Auxiliary Building elevator shaft smoke/heat hole has been evaluated and determined to be an acceptable minor leakage pathway. Therefore, spent fuel pool area integrity is maintained, and the required Enclosure Building Filtration Train is OPERABLE, when the elevator shaft smoke/heat hole is open. 2-HV-171, Spent Fuel Pool Area Exhaust Damper, is not an acceptable bypass leakage path and must remain closed when necessary to maintain spent fuel pool area integrity.

The laboratory testing requirement for the charcoal sample to have a removal efficiency of $\geq$ 95% is more conservative than the elemental and organic iodine removal efficiencies of 90% and 70%, respectively, assumed in the DBA analyses for the EBFS charcoal adsorbers in the Millstone Unit 2 Final Safety Analysis Report. A removal efficiency acceptance criteria of $\geq$ 95% will ensure the charcoal has the capability to perform its intended safety function throughout the length of an operating cycle.

3/4.9.16 SHIELDED CASK

The limitations of this specification and 3/4.9.15 ensure that in the event of a shielded cask drop accident the doses from ruptured fuel assemblies will be within the assumptions of the safety analyses.

3/4.9.17 SPENT FUEL POOL BORON CONCENTRATION

The limitations of this specification ensures that sufficient boron is present to maintain spent fuel pool $K_{eff} \leq 0.95$ under accident conditions.

Postulated accident conditions which could cause an increase in spent fuel pool reactivity are: a single dropped or mis-loaded fuel assembly, a single dropped or mis-loaded Consolidated Fuel Storage Box, or a shielded cask drop onto the storage racks. A spent fuel pool soluble boron concentration of 1400 ppm is sufficient to ensure $K_{eff} \leq 0.95$ under these postulated accident conditions. The required spent fuel pool soluble boron concentration of $\geq$ 1720 ppm conservatively bounds the required 1400 ppm. The ACTION statement ensures that if the soluble boron concentration falls below the required amount, that fuel movement or shielded cask movement is stopped, until the boron concentration is restored to within limits.

An additional basis of this LCO is to establish 1720 ppm as the minimum spent fuel pool soluble boron concentration which is sufficient to ensure that the design basis value of 600 ppm soluble boron is not reached due to a postulated spent fuel pool boron dilution event. As part of the spent fuel pool criticality design, a spent fuel pool soluble boron concentration of 600 ppm is sufficient to ensure $K_{eff} \leq 0.95$, provided all fuel is stored consistent with LCO requirements. By maintaining the spent fuel pool soluble boron concentration $\geq$ 1720 ppm, sufficient time is provided to allow the operators to detect a boron dilution event, and terminate the event, prior to the spent fuel pool being diluted below 600 ppm. In the unlikely event that the spent fuel pool soluble boron concentration is decreased to 0 ppm, K_{eff} will be maintained < 1.00, provided all fuel is stored consistent with LCO requirements. The ACTION statement ensures that if the soluble boron concentration falls below the required amount, that immediate action is taken to restore the soluble boron concentration to within limits, and that fuel movement or shielded cask movement is stopped. Fuel movement and shielded cask movement is stopped to prevent the possibility of creating an accident condition at the same time that the minimum soluble boron is below limits for a potential boron dilution event.

The surveillance of the spent fuel pool boron concentration within 24 hours of fuel movement, consolidated fuel movement, or cask movement over the cask laydown area, verifies that the boron concentration is within limits just prior to the movement. The 7 day surveillance interval frequency is sufficient since no deliberate major replenishment of pool water is expected to take place over this short period of time.

3/4.9.18 SPENT FUEL POOL - STORAGE

The limitations described by Figures 3.9-1a, 3.9-1b, and 3.9-3 ensure that the reactivity of fuel assemblies and consolidated fuel storage boxes, introduced into the Region C spent fuel racks, are conservatively within the assumptions of the safety analysis.

The limitations described by Figure 3.9-4 ensure that the reactivity of the fuel assemblies, introduced into the Region A spent fuel racks, are conservatively within the assumptions of the safety analysis.

3/4.9.19 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity condition of the Region B storage racks and spent fuel pool K_{eff} will remain less than or equal to 0.95.

The Cell Blocking Devices in the 4th location of the Region B storage racks are designed to prevent inadvertent placement and/or storage in the blocked locations. The blocked location remains empty, or a Batch B fuel assembly may be stored in the blocked location, to maintain reactivity control for fuel assembly storage in any adjacent locations. Region B (non-cell blocker locations) is designed for the storage of new assemblies in the spent fuel pool, and for fuel assemblies which have not sustained sufficient burnup to be stored in Region A or Region C.

This LCO is not applicable during the initial installation of Batch B fuel assemblies in the cell blocker locations of Region B. This is acceptable because only Batch B fuel assemblies will be moved during the initial installation of Batch B fuel assemblies under the Region B cell blockers. Batch B fuel assemblies are qualified for storage in any spent fuel pool storage rack location, hence a fuel misloading event which causes a reactivity consequence is not credible. This exception is valid only during the initial installation of Batch B fuel assemblies in the cell blocker locations.

3/4.9.20 SPENT FUEL POOL - CONSOLIDATION

The limitations of these specifications ensure that the decay heat rates and radioactive inventory of the candidate fuel assemblies for consolidation are conservatively within the assumptions of the safety analysis.

Attachment 5

Millstone Power Station, Unit No. 2

Technical Specifications Change Request 2-10-01
Fuel Pool Requirements
Criticality Analysis

Millstone Unit 2 Spent Fuel Pool Criticality Analysis with Soluble Boron Credit

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1 Introduction

1.1 Objective

This report presents the results of criticality analyses for the Millstone Unit 2 spent fuel storage racks. The physical description of the spent fuel storage racks is provided in Reference 1. The primary objectives of this analysis are as follows:

1. to employ Soluble Boron Credit in establishing the design/storage basis for the spent fuel storage racks.

2. to establish the maximum radially-averaged fresh fuel enrichment limit in Region B. Region B will be employed to store fresh and burned fuel assemblies in a 3/4 (three fresh/one burned) storage configuration. The burned fuel assembly is a Combustion Engineering Batch B fuel assembly with an maximum initial enrichment less than or equal to 2.36 w/o U-235 and a minimum assembly burnup greater than or equal to 22,300 MWD/MT. The fresh fuel enrichment limit will be based upon this storage configuration along with a degraded Boraflex representation. All Boraflex panels are assumed to contain 5.65 inch gaps. The gaps surrounding individual assemblies are coplanar. The gaps are modeled along the midplane of the active fuel height and are "staggered" in adjacent locations by four (4) inches. In addition, the Boraflex was modeled with a reduced B^{10} loading equal to 0.025 g B^{10} per square centimeter.

3. to establish assembly burnup versus initial enrichment limits for the fuel assemblies stored in Regions A. Region A will be employed to store low burnup fuel assemblies which have not achieved their expected discharge burnup (e.g., once burned fuel assemblies.). Region A also contains Boraflex and the Boraflex model used in Region A is exactly the same as employed for Region B. Also, to investigate the removal of two Boraflex poison boxes from any module in Region A for future in-service testing.

4. to establish assembly burnup versus initial enrichment limits for the fuel assemblies stored in Regions C. Region C will be employed to store normally discharged fuel assemblies in a high density configuration (4/4). In Region C the burnup versus initial enrichment storage curve will be generated with and with out three poison RODLETS that are located in the guide tubes (one RODLET per guide tube) of the fuel assembly. In addition, Region C may be employed to store Consolidated Fuel Storage Boxes (CFSB). The burnup versus initial enrichment limits for fuel rods stored in a CFSB were previously evaluated. This analysis will not evaluate these limits except to state that the existing limits are based upon a more conservative constraint. The burnup versus initial enrichment limits for the

CFSB were generated based upon a maximum k-effective value, including all biases and uncertainties, less than or equal to 0.95 at zero ppm.

5. to determine the soluble boron required to maintain k-effective less than or equal to 0.95, including all biases and uncertainties, assuming the most limiting reactivity accident.

The methodology employed in this analysis for soluble boron credit is analogous to that of Reference 17 and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 3[1]. Reference 17 was reviewed and approved by the US NRC. The methodology employed in this analysis and in Reference 17 employs axially distributed burnups to represent discharged fuel assemblies.

1.2 Design Criteria

The design criteria are consistent with GDC 62, Reference 4, and NRC guidance given in Reference 5. Section 2.0 describes the analysis methods including a description of the computer codes used to perform the criticality safety analysis. A brief summary of the analysis approach and criteria follows.

1. Determine the fresh and spent fuel storage configuration of the spent fuel pool using no soluble boron conditions such that the 95/95 upper tolerance limit value of k_{eff} for the storage pool, including applicable biases and uncertainties, is less than unity.

2. Next, using the resulting storage configurations from the previous step, calculate the spent fuel rack effective multiplication factor with the chosen concentration of spent fuel pool soluble boron present. Then calculate the sum of: (a) the latter multiplication factor, (b) the reactivity uncertainty associated with fuel assembly and storage rack tolerances, and (c) the biases and other uncertainties required to determine the final 95/95 confidence level effective multiplication factor and show that at the chosen concentration of soluble boron, the system maintains the overall effective multiplication factor less than or equal to 0.95.

[1] Note that only the criteria from Reference 3 was used and not the methodology. Thus the restrictions of Reference 18 do not apply.

3. Determine the increase in reactivity caused by postulated accidents and the corresponding additional amount of soluble boron needed to offset these reactivity increases.

An alternative form of expressing the soluble boron requirements is given in Reference 3. The final soluble boron requirement is determined from the following summation:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA} \qquad \textbf{Equation 1}$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm),

$SBC_{95/95}$ = soluble boron requirement for 95/95 $k_{eff} \leq 0.95$ (ppm),

SBC_{RE} =soluble boron required for reactivity equivalencing methodologies (ppm),

SBC_{PA} = soluble boron required for $k_{eff} \leq 0.95$ under accident conditions (ppm).

For purposes of the analyses contained herein, minimum burnup limits established for fuel assemblies to be stored in the storage racks do include burnup credit established in a manner which takes into account conservative approximations to the operating history of the fuel assemblies. Variables such as the axial burnup profile as well as the axial profile of moderator and fuel temperatures have been factored into the analyses.

1.3 Design Approach

The design input employed in this analysis was directly obtained from Reference 1. The Soluble Boron Credit Methodology provides additional reactivity margin in the spent fuel storage analyses which may then be used to implement added flexibility in storage criteria and, for example, to implement degraded Boraflex modeling. The Millstone Unit 2 spent fuel pool storage racks can be categorized into three regions – Region A, B, and C. Regions A and B contain Boraflex which is conservatively analyzed in this analysis. Region C does not contain Boraflex. The Boraflex was modeled with 5.65 inch gaps staggered about the midplane of the active fuel height. The gaps surrounding a fuel assembly are completely aligned. In adjacent locations the aligned gaps are staggered 4 inches away from each other. Therefore, all of the Boraflex gaps are within a 15.3 inch layer, which is centered about the midplane of the active fuel height. This Boraflex gap model is conservative relative to a Boraflex model which randomly distributes the gaps over the entire length of the Boraflex panel.

The selection of design basis fuel assembly types was based on an evaluation of the variety of fuel assemblies employed in the reactor to date and selecting the most reactive type for a given evaluation. The candidate fuel assembly types include the Westinghouse Standard, Siemens and Combustion Engineering 14x14 fuel assembly designs. The Westinghouse Standard fuel assembly has been evaluated to be the design basis fuel assembly to represent burned fuel assemblies stored in Region C while the Siemens assembly design has been evaluated to be the design basis fuel assembly to represent fresh fuel and low burnup fuel assemblies in Regions B and A respectively. The Combustion Engineering assembly design was used as the design basis fuel assembly to represent the burned fuel assemblies located under the Region B cell blocker locations, since this is the only design allowed under the cell blocker locations. The most reactive fuel assembly design will be employed for each subregion of the spent fuel pool. Fuel assembly designs used in these calculations do not credit burnable absorbers or grids. In addition, the most reactive spent fuel pool temperature will be employed in each subregion such that the analysis results are valid over the entire spent fuel temperature range. The spent fuel temperature range is limited to temperatures less than or equal to 150 degrees Fahrenheit. Thus, the analysis results will be valid for spent fuel pool temperatures less than or equal to 150 degrees Fahrenheit. Temperatures greater than 150 degrees Fahrenheit are considered in the accident analyses in which credit for soluble boron is allowed.

The reactivity characteristics of the storage racks were evaluated using infinite lattice analyses; this environment was employed in the evaluation of the burnup limits versus initial enrichment as well as the evaluation of physical tolerances and uncertainties. A full spent fuel pool model was also employed to evaluate soluble boron worth, the reactivity worth of postulated accidents, and the multiplication factor for the zero soluble boron condition.

1.4 Methodology

This section describes the analysis methodology employed to assure the criticality safety of the spent fuel pools and to define limits placed on fresh and spent fuel storage in the three regions of the Millstone Unit 2 spent fuel pool. The analysis methodology employs: (1) SCALE-PC, a personal computer version of the SCALE-4.3 code system, as documented in Reference 6, with the updated SCALE-4.3 version of the 44 group ENDF/B-V neutron cross section library, and (2) the two-dimensional integral transport code DIT, Reference 7, with an ENDF/B-VI neutron cross section library.

SCALE-PC is used for calculations involving infinite arrays of storage cells in Regions A, B and C racks. In addition, it is employed in a full pool representation of the storage racks to evaluate soluble boron worth and postulated accidents.

SCALE-PC modules employed in both the benchmarking analyses and the spent fuel storage rack analyses include the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENOV.a. All references to KENO in the text to follow should be interpreted as referring to the KENO V.a module.

The DIT code is used for simulation of in-reactor fuel assembly depletion. The following sections describe the application of these codes in more detail.

1.4.1 SCALE-PC

The SCALE system was developed for the Nuclear Regulatory Commission to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system which runs on specific classes of personal computers.

1.4.2 Validation of SCALE-PC

Validation of SCALE-PC for purposes of fuel storage rack analyses is based on the analysis of selected critical experiments from two experimental programs. The first program is the Babcock & Wilcox (B&W) experiments carried out in support of Close Proximity Storage of Power Reactor Fuel, Reference 8. The second program is the Pacific Northwest Laboratory (PNL) Program carried out in support of the design of Fuel Shipping and Storage Configurations; the experiments of current interest to this effort are documented in Reference 9. Reference 10, as well as several of the relevant thermal experiment evaluations in Reference 11, were found to be useful in updating pertinent experimental data for the PNL experiments.

Nineteen experimental configurations were selected from the B&W experimental program; these consisted of the following experimental cores: Core X, the seven measured configurations of Core XI, Cores XII through XXI, and Core XIIIa. These analyses employed measured critical data, rather than the extrapolated configurations to a fixed critical water height reported in Reference 8, so as to avoid introducing possible biases or added uncertainties associated with the extrapolation techniques. In addition to the active fuel region of the core, the full environment of the latter region, including the dry fuel above the critical water height, was represented explicitly in the analyses.

The B&W group of experimental configurations employed variable spacing between individual rod clusters in the nominal 3 x 3 array. In addition, the effects of placing either SS-304 or Borated Aluminum plates of different boron contents in the water channels between rod clusters were measured. Table 1.4-1 summarizes the results of these analyses.

Eleven experimental configurations were selected from the PNL experimental program. These experiments included unpoisoned uniform arrays of fuel pins and 2 x 2 arrays of rod clusters with and without interposed SS-304 or B/Al plates of different blacknesses. As in the case of the B&W experiments, the full environment of the active fuel region was represented explicitly. Table 1.4-2 summarizes the results of these analyses.

The approach employed for a determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of Reference 12. For a given KENO calculated value of k_{eff} and associated one sigma uncertainty, the magnitude of $k_{95/95}$ is computed by the following equation; by this definition, there is a 95 percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than $k_{95/95}$.

$$k_{95/95} = k_{KENO} + \Delta k_B + M_{95/95}\left(\sigma_m^2 + \sigma_{KENO}^2\right)^{\frac{1}{2}} \qquad \textbf{Equation 2}$$

where:

k_{KENO} is the KENO multiplication factor of interest,

Δk_B is the mean calculational method bias,

$M_{95/95}$ is the 95/95 multiplier appropriate to the degrees of freedom for the number of validation analyses,

σ_m^2 is the mean calculational method variance deduced from the validation analyses,

σ_{KENO}^2 is the square of the KENO standard deviation.

The equation for the mean calculational methods bias is as follows;

$$\Delta k_B = \frac{1}{n}\sum_{i=1}^{n}\left(1 - k_i\right), \qquad \textbf{Equation 3}$$

where:

k_i is the i^{th} value of the multiplication factor for the validation lattices of interest.
$M_{95/95}$ is obtained from the tables of Reference 13.

The equation for the mean calculational variance of the relevant validating multiplication factors is as follows.

$$\sigma_m^2 = \left[\frac{n\sum_1^n (k_i - k_{ave})^2 \sigma_i^{-2}}{(n-1)\sum_1^n \sigma_i^{-2}} \right] - \sigma_{ave}^2 \qquad \textbf{Equation 4}$$

Where k_{ave} is given by the following equation.

$$k_{ave} = \frac{\sum_1^n k_i \sigma_i^{-2}}{\sum_1^n \sigma_i^{-2}} \qquad \textbf{Equation 5}$$

σ_{ave}^2 is given by the following equation.

$$\sigma_{ave}^2 = \frac{\sum_1^n \sigma_i^2 G_i}{\sum_1^n G_i} \qquad \textbf{Equation 6}$$

Where G_i is the number of generations.

For purposes of this bias evaluation, the data points of Table 1.4-1 and Table 1.4-2 are pooled into a single group. With this approach, the mean calculational methods bias, Δk_B, and the mean calculational variance, $(\sigma_m)^2$, calculated by equations given above, are determined to be 0.00259 and $(0.00288)^2$, respectively. The magnitude of $M_{95/95}$ is deduced from Reference 13 for the total number of pooled data points, 30.

The magnitude of $k_{95/95}$ is given by the following equation for SCALE 4.3 KENO analyses employing the 44 group ENDF/B-V neutron cross section library and for analyses where these experiments are a suitable basis for assessing the methods bias and calculational variance.

$$k_{95/95} = k_{KENO} + 0.00259 + 2.22\left[(0.00288)^2 + \sigma^2_{KENO}\right]^{\frac{1}{2}} \qquad \textbf{Equation 7}$$

Based on the above analyses, the mean calculational bias, the mean calculational variance, and the 95/95 confidence level multiplier are deduced as 0.00259, $(0.00288)^2$, and 2.22, respectively.

1.4.3 Application to Fuel Storage Pool Calculations

As noted above, the CSAS25 control module was employed to execute the functional modules within SCALE-PC. The CSAS25 control module was used to analyze either infinite arrays of single or multiple storage cells or the full spent storage pool.

Standard material compositions were employed in the SCALE-PC analyses consistent with those of Reference 1; these data are listed in Table 1.4-3. For fresh fuel conditions, the fuel nuclide number densities were derived within the CSAS25 module using input consistent with the data of Table 1.4-3. For burned fuel representations, the fuel isotopics were derived from the DIT code as described below.

1.4.4 The DIT Code

The DIT (Discrete Integral Transport) code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The neutron transport equations are solved in integral form within each pin cell. The cells retain full heterogeneity throughout the discrete integral transport calculations. The multigroup spectra are coupled between cells through the use of multigroup interface currents. The angular dependence of the neutron flux is approximated at cell boundaries by a pair of second order Legendre polynomials. Anisotropic scattering within the cells, together with the anisotropic current coupling between cells, provide an accurate representation of the flux gradients between dissimilar cells.

The multigroup cross sections are based on the Evaluated Nuclear Data File Version 6 (ENDF/B-VI). Cross sections have been collapsed into an 89 group structure which is used in the assembly spectrum calculation. Following the multigroup spectrum calculation, the region-wise cross sections within each heterogeneous cell are collapsed to a few groups (usually 4 broad groups), for use in the assembly flux calculation. A B1 assembly leakage correction is performed to modify the spectrum according to the assembly in- or out-leakage. Following the flux calculation, a depletion step is performed to generate a set of region-wise isotopic concentrations at the end of a burnup interval. An extensive set of depletion chains are available, containing 33 actinide nuclides in the thorium, uranium and plutonium chains, 171 fission products, the gadolinium, erbium and boron depletable absorbers, and all structural nuclides. The spectrum-depletion sequence of calculations is repeated over the life of the fuel assembly. Several restart capabilities provide the temperature, density, and boron concentration dependencies needed for three dimensional calculations with full thermal-hydraulic feedback effects.

The DIT code and its cross section library are employed in the design of initial and reload cores and have been extensively benchmarked against operating reactor history and test data.

For the purpose of spent fuel pool criticality analysis calculations, the DIT code is used to generate the detailed fuel isotopic concentrations as a function of fuel burnup and initial feed enrichment. Each selected set of fuel isotopics is equivalenced to a reduced set of burned fuel isotopics at specified time points after discharge. The latter burned fuel representation includes the following nuclides: ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{149}Sm, ^{16}O, and ^{10}B. The DIT code lists the Samarium-149 isotopics for ^{149}Sm and ^{149D}Sm (a metastable isomer). Since ^{149}Sm is a stable isotope, the concentration of this Samarium isotope is the sum of the individual concentration of these two isomers.

The isotopic number densities from the DIT calculation are based upon Cell average values. The input to KENO calculations require that the number densities be specified for the fuel pellet. Therefore, the number densities from the DIT calculations are scaled by the ratio of area of the cell to the area of the fuel pellet for use in the KENO calculations. The concentration of ^{10}B is determined by reactivity equivalencing a given DIT cell calculation with a corresponding KENO cell calculation to within the KENO one sigma uncertainty level.

Table 1.4-1
Summary of Calculational Results for
Cores X Through XXI of the B&W Close Proximity Experiments

Core	Run No.	KENO K_{eff}	Plate Type[2]	Spacing[3]
X	2348	0.99610 ± 0.00084	none	3
XI	2355	1.00049 ± 0.00080	SS-304	1
XI	2359	0.99884 ± 0.00077	SS-304	1
XI	2360	1.00315 ± 0.00081	SS-304	1
XI	2361	0.99831 ± 0.00080	SS-304	1
XI	2362	1.00060 ± 0.00078	SS-304	1
XI	2363	0.99957 ± 0.00078	SS-304	1
XI	2364	1.00246 ± 0.00080	SS-304	1
XII	2370	0.99990 ± 0.00082	SS-304	2
XIII	2378	0.99754 ± 0.00089	B/Al	1
XIIIA	2423	0.99575 ± 0.00087	B/Al	1
XIV	2384	0.99465 ± 0.00086	B/Al	1
XV	2388	0.99158 ± 0.00084	B/Al	1
XVI	2396	0.99230 ± 0.00088	B/Al	2
XVII	2402	0.99478 ± 0.00079	B/Al	1
XVIII	2407	0.99440 ± 0.00083	B/Al	2
XIX	2411	0.99821 ± 0.00081	B/Al	1
XX	2414	0.99498 ± 0.00082	B/Al	2
XXI	2420	0.99318 ± 0.00094	B/Al	3

[2] Entry indicates metal separating unit assemblies.
[3] Entry indicates spacing between unit assemblies in units of fuel rod pitch.

Table 1.4-2
Summary of Calculational Results for Selected Experimental PNL Lattices, Fuel Shipping and Storage Configurations

Exp't. No.	K_{eff}	Comments
043	0.99787 ± 0.00106	Uniform rectangular array, no poison
044	1.00104 ± 0.00102	"
045	0.99955 ± 0.00101	"
046	0.99960 ± 0.00103	"
061	0.99792 ± 0.00099	2 x 2 array of rod clusters, no poison
062	0.99628 ± 0.00096	"
064	0.99696 ± 0.00103	2 x 2 array of rod clusters, 0.302 cm thick SS-304 cross
071	0.99970 ± 0.00101	2 x 2 array of rod clusters, 0.485 cm thick SS-304 cross
079	0.99463 ± 0.00102	2 x 2 array of rod clusters, cross of 0.3666 g boron/cm^2
087	0.99423 ± 0.00099	2 x 2 array of rod clusters, cross of 0.1639 g boron/cm^2
093	0.99787 ± 0.00098	2 x 2 array of rod clusters, cross of 0.1425 g boron/cm^2

Table 1.4-3
Standard Material Compositions Employed in Criticality Analysis for Millstone Unit 2 Spent Fuel Storage Rack

Material	Element	Weight Fraction
Zircaloy-4	Zr	0.9829
Den.= 6.56 g/cc	Sn	0.0140
	Fe	0.0021
	Cr	0.0010
Water	SCALE Standard Composition Library Density=1.0018 g/cc	
Fresh UO_2	SCALE Standard Composition Library Stack Density = 0.9535	
SS304	SCALE Standard Composition Library	
Boraflex	N_{B10} = 5.38226E-03 atom/b·cm	
Areal Density =	N_{B11} = 2.17982E-02 atom/b·cm	
0.025 g B10/cm^2	N_C = 1.61955E-02 atom/b·cm	
	N_{Si} = 8.93104E-03 atom/b·cm	
	N_O = 1.40781E-02 atom/b·cm	
Borated SS304	B	2.0 wt %
Den. = 7.76 g/cc	SS304	98.0 wt %

1.5 Assumptions

1. The Westinghouse Standard fuel assembly was modeled with a fuel density equal to 10.4504 g/cc (95.35 % of theoretical density). This is a conservative value relative to the value listed in Table 1 of Reference 1. In addition, this fuel assembly was modeled with no credit for grids and other structural materials.

2. The analysis conservatively assumed a 0.1 inch gap between all storage modules. In reality the gaps between modules are a couple of inches wide. Therefore, the KENO results for the entire spent fuel pool are conservatively calculated.

1.6 Analysis Results

The primary objectives of this analysis were accomplished; a summary of the results is as follows.

1. Soluble boron credit methodology was employed to calculate a k_{eff} value of 0.97086 for the entire spent fuel pool at zero soluble boron. The allowance for applicable biases and uncertainties was deduced to be 0.02499 delta-k; thus, the 95/95 upper tolerance limit value of k_{eff} was deduced to be 0.99585. The total soluble boron requirement for achieving a 95/95 value of $k_{eff} \leq 0.95$ was deduced to be the summation of the following three terms: $SBC_{95/95}$ = 383 ppm, SBC_{RE} = 180 ppm, and SBC_{PA} = 790 ppm for a total of 1353 ppm. The ^{10}B atom percent used in this analysis is 19.9 a/o. The soluble boron concentration required for a ^{10}B atom percent equal to 19.7 a/o is $SBC_{95/95}$ = 387 ppm, SBC_{RE} = 182 ppm, and SBC_{PA} = 1367 ppm. Note that this soluble boron concentration includes an allowance for 5 % burnup uncertainty.

2. The design basis fuel assembly for the fresh fuel assemblies stored in Region B storage racks was taken to be a conservative representation of the Siemens 14 x 14 fuel assembly having a maximum radially averaged enrichment less than or equal to 4.85 wt% ^{235}U and no burnable poisons. This enrichment limit is based upon a 3/4 (three fresh/one burned) storage configuration. The burned fuel assembly is a Combustion Engineering Batch B fuel assembly with a maximum initial enrichment less than or equal to 2.36 w/o U-235 and a minimum assembly burnup greater than or equal to 22,300 MWD/T. The Batch B fuel assemblies were simulated in Region B cell locations which are presently blocked. The fresh fuel enrichment limit is based upon this storage configuration along with a degraded Boraflex representation. All Boraflex panels are assumed to contain 5.65 inch gaps. The gaps surrounding individual assemblies are coplanar. The gaps are modeled along the midplane of the active fuel height and are "staggered" in adjacent locations by four (4) inches. In addition, the Boraflex was modeled with a reduced B^{10} loading equal to 0.025 g B^{10}

per square centimeter. Note that the fuel stack density was modeled as 95.35 % of theoretical density.

3. The design basis fuel assembly for the low burnup fuel assemblies stored in the Region A storage racks was taken to be a conservative representation of the Siemens 14 x 14 fuel assembly having a maximum radially averaged initial enrichment equal to 3.64 wt% ^{235}U at zero burnup. The burnup versus initial enrichment limits for storage in Region A are given in Figure 4.2-1.

4. The design basis fuel assembly for the burned fuel assemblies stored in Region C storage racks was taken to be a conservative approximation to the Westinghouse Standard 14 x 14 fuel assembly. This conservative approximation to the burned fuel assembly envelopes the characteristics of all burned fuel assemblies currently stored in the spent fuel pool. This design basis burned fuel assembly was represented by a four-node axial representation of the assembly burnup and applicable fuel and moderator temperatures. Figure 4.2-2 contains the burnup versus initial enrichment limits for storage in Region C without the use of poison RODLETS. Figure 4.2-3 contains the burnup versus initial enrichment limits for storage in Region C with a poison RODLET located in three guide tube locations (one in the center and two in diagonally opposite locations.)

5. Region C may also be employed to store Consolidated Fuel Storage Boxes (CFSB). The burnup versus initial enrichment limits for fuel rods stored in a CFSB were previously evaluated and are given in Figure 3.9-3 of Reference 1. This analysis did not evaluate these limits except to state that the existing limits were derived based upon a more conservative constraint; the burnup versus initial enrichment limits for the CFSB were derived based upon a maximum K-effective value, including all biases and uncertainties, less than or equal to 0.95 at a soluble boron concentration of zero ppm. Therefore, the burnup versus initial enrichment limits for the CFSB in the existing Technical Specifications are judged to be conservative and remain applicable.

The following accidents were considered in this analysis:

- Misloaded fresh fuel assembly into either Region A, B or C
- Misloaded fresh fuel assembly just outside the storage racks.
- Misloaded fresh fuel assembly near the fuel elevator which contains a fresh fuel assembly
- Heavy load accident in Regions A, B, and C
- Dropped fresh fuel assembly on top of the storage racks.
- Seismic Event which would reduce the intramodule gaps
- Spent fuel pool temperature greater than 150 degrees Fahrenheit

The most limiting accident condition was determined to be the heavy load accident in Region B . The heavy load accident was simulated by collapsing the poison boxes onto the outside envelope of the fuel assemblies and the rack onto the outside envelope of the collapsed poison boxes. The heavy load accident in Region B requires 790 ppm to mitigate the event.

2 Design Input

As noted in the Introduction Section, the Millstone Unit 2 spent fuel storage pool configuration and the individual storage racks as analyzed herein are consistent with Reference 1. This section provides a brief description of the spent fuel storage racks with the objective of establishing a basis for the analytical model employed in the criticality analyses described in Section 3.1.

2.1 Spent Fuel Pool Storage Configuration Description

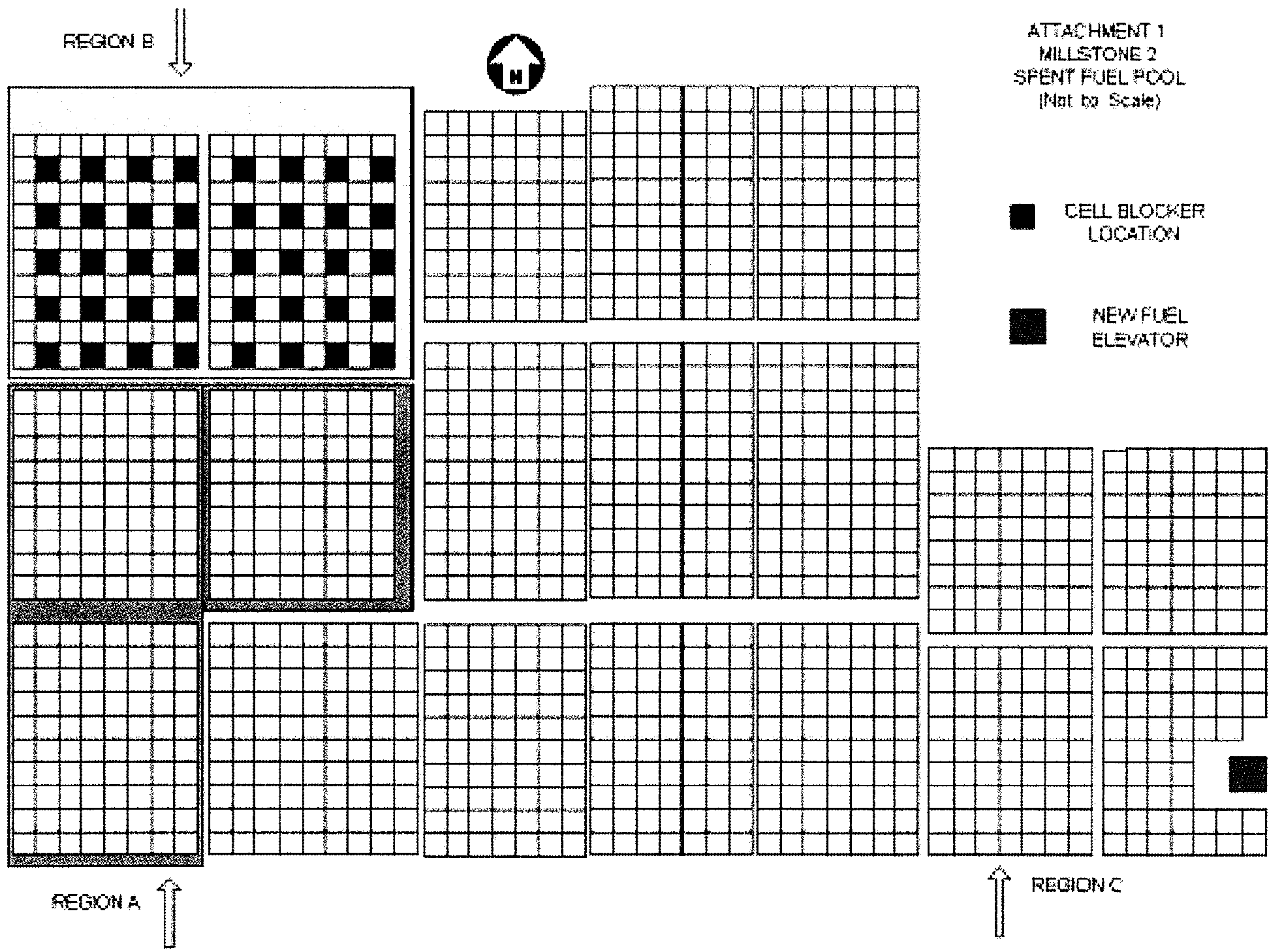
The existing Millstone Unit 2 spent fuel pool and the fuel storage rack types and orientation are illustrated in Figure 2.1-1. The fresh and spent fuel assembly storage area in the pool is divided into three regions; Region A, Region B, and Region C. Regions A and B employ a flux-trap design and a Boraflex panel in each storage cell for reactivity control. Region A and B are primarily employed to store fuel assemblies which have not yet achieved their expected discharge burnup (e.g. fresh fuel assemblies and low burnup fuel assemblies). Region C is employed to store discharged fuel assemblies (both consolidated and intact) which have achieved their expected discharge burnup. Region C does not contain any poison material for reactivity control.

Region A consists of one 8 x 10 module and two 8 x 9 modules with a nominal center-to-center spacing equal to 9.80 inches. The Region A storage racks can store up to 224 spent fuel assemblies.

Region B consists of two 8 x 10 modules with cell blockers presently installed in 40 locations. The cell blockers are depicted in Figure 2.1-1 as darkly colored squares. One of the purposes of this analysis is to demonstrate that Batch B fuel assemblies with an initial enrichment less than or equal to 2.36 w/o U-235 and a minimum assembly burnup greater than or equal to 22,300 MWD/MT may be stored in these blocked locations.

Region C consists of fourteen modules with a nominal center-to-center spacing equal to 9.0 inches. There are two 8 x 7 modules, three 9 x 7 modules (with one missing 10 storage locations for the fuel elevator function), five 10 x 7 modules, three 11 x7 modules, and one 19 x 9 module. The Region C modules contain a total of 962 storage locations and are capable of storing both consolidated and intact fuel assemblies.

Figure 2.1-1
Spent Fuel Pool, General Arrangement



2.2 Individual Storage Rack Type Descriptions

Subsequent sections describe the individual storage racks in greater detail.

2.2.1 Region A and B Storage Racks.

Each Region A or B storage cell is centered on a nominal 9.8 inch pitch and consists of an open stainless steel box. The stainless steel box has an nominal inside dimension equal to 9.665 inches and is 0.135 inches thick. Centered in the open stainless steel box is an open poison box which contains the Boraflex material. The inner poison box was constructed with an inner layer of stainless steel, a middle layer of Boraflex, and an outer layer of stainless steel. The inner and outer layer of stainless steel are 0.029 inches thick. The Boraflex layer is 0.110 inches thick and is separated from the stainless steel by a 0.003 inch void on each side. Overall, the inner box has a nominal inside dimension equal to 8.71 inches. The Boraflex material was originally manufactured with an areal density equal to 0.033 g-B^{10} per square centimeter. In this analysis the Boraflex material is assumed to have an areal density equal to 0.025 g-B^{10} per square centimeter, which represents approximately a 25 % reduction from the manufactured areal density. Table 2.2-1 contains a full listing of the dimensional data employed to model the Regions A and B storage racks. This data was obtained from Figure 1 of Reference 1.

2.2.2 Region C Storage Racks

Each Region C storage cell consist of an open stainless steel box with an nominal inside dimension equal to 8.865 inches and a nominal thickness equal to 0.135 inches. The nominal cell-to-cell spacing is 9.00 inches. The pertinent dimensions of the constituent materials for the Region C storage cells are also summarized in Table 2.2-2. This data was obtained from Page 2 of Reference 1.

2.2.3 Poison RODLETS

Poison RODLETS may be employed in Region C to lower the burnup required to store discharged fuel assemblies in this region. The RODLETS were modeled in this analysis as cylindrical objects with an outside diameter equal to 0.87 inches. When RODLETS are used, three RODLETS are required per fuel assembly, with 1 RODLET in the center guide tube, and the other 2 RODLETS stored in diagonally opposite guide tubes. The RODLETS were manufactured from borated stainless steel with a nominal boron concentration equal to 2.0 weight percent natural boron. The RODLETS were modeled in this analysis as starting three (3) inches above the bottom of the active fuel height. This data was obtained from Page 2 of Reference 1. Fuel assemblies containing RODLETS may be stored in any rotational orientation.

2.2.4 Limiting Axial Burnup Distribution Profile

Input to this analysis is the limiting axial burnup profile data provided in the DOE Topical Report as documented in Reference 14. The burnup profile in the DOE Topical Report is based on a database of 3169 axial-burnup profiles for PWR fuel assemblies compiled by Yankee Atomic. This profile is derived from the burnups calculated by utilities or vendors based on core-follow calculations and in-core measurement data. The axial burnup profile in the DOE report is based on the most limiting axial burnup shape found in the database. The four zone model is constructed based on this limiting axial burnup profile. Table 3.3-1 (third column) contains the limiting relative axial burnup shape as applied in this analysis.

The zone moderator and fuel temperatures for the four zone model were taken from typical Millstone Unit 2 midcycle reactor core axial temperature profiles provided in Reference 16.

Table 2.2-1
Region A and B Storage Rack Cell Dimensions

Description	Design Dimensions
Cell Pitch, cm.(in.)	24.892 ± 0.2286 (9.80 ± 0.09)
Cell ID, cm.(in.)	22.1234 ± 0.127 (8.710 ± 0.050)
Boraflex Plate Thickness, cm. (in.)	0.2794 ± 0.01778 (0.110 ± 0.007)
SS cover sheet thickness, cm.(in.)	0.07366 (0.029)
Stainless Steel Wall Thickness, cm.(in.)	0.3429 ± 0.0304 (0.135 ± 0.012)

Table 2.2-2
Region C Storage Rack Cell Dimensions

Description	Design Dimensions
Cell Pitch, cm.(in.)	22.86 (9.0)
BSS Rodlet OD cm. (in.)	2.2098 ± 0.0381 (0.87 ± 0.015)
Stainless Steel Wall Thickness, cm.(in.)	0.3429 ± 0.03048 (0.135 ± 0.012)

3 Analysis

3.1 KENO Models

The purpose of this section is to describe the models employed in KENO to represent either arrays of different types of storage cells or the full spent pool storage configuration. The current Millstone 2 spent fuel pool consists of 3 storage regions A, B and C:

3.1.1 Region B

The Region B storage cell is modeled in KENO as a square cell that has a side dimension of 9.8 inch. The open stainless steel box extends inward for a distance equal to half of the nominal thickness of 0.135 inch and has an nominal inside dimension equal to 9.665 inch. The inner and outer layer of stainless steel on the poison box are 0.029 inches thick. The Boraflex layer is 0.110 inches thick and is separated from the stainless steel by a 0.003 inch void on each side. The inner box has a nominal inside dimension equal to 8.71 inches. These rack dimensions are shown in Figure 2.1-1. The Region B storage cell is a composite of fuel assembly cell, Boraflex flux trap, stainless steel assembly holder, and stainless steel rack. The Siemens (current) and C-E (original core) fuel types are displayed in Figure 3.2-1. A detailed summary of geometric specifications as used for KENO input to represent fuel assembly are shown in Table 3.2-1.

The Boraflex degradation model simulates a Boraflex-void region 5.65 inch in height in every Boraflex panel, that is staggered either 2 inch above or below the fuel centerline. For each available adjacent storage cell the Boraflex-void location is mirrored in the axial direction. This pattern is shown in Figure 3.2-2 that was captured from KENO plotter output. Figure 3.2-2 shows an axial elevation where Boraflex is present (shown in yellow) surrounding 2 fuel assemblies, and where Boraflex is missing (void–light gray) surrounding 2 other assemblies.

3.1.2 Region A

The Region A storage cell is modeled in KENO as a square cell that has a side dimension of 9.8 inch. The open stainless steel box extends inward for a distance equal to half of the nominal thickness of 0.135 inch and has an nominal inside dimension equal to 9.665 inch. The inner and outer layer of stainless steel on the poison box are 0.029 inches thick. The Boraflex layer is 0.110 inches thick and is separated from the stainless steel by a 0.003 inch void on each side. The inner box has a nominal inside dimension equal to 8.71 inches. These rack dimensions are shown in Figure 3.1-2. A 14x14 array of fuel for Siemens (current) fuel type is displayed in Figure 3.2-1. Geometric specifications for Siemens (current) fuel type are listed in Table 3.2-1.

The Boraflex degradation model simulates a Boraflex-void region 5.65 inch in height in every Boraflex panel that is staggered either 2 inch above or below the fuel centerline. The Boraflex degradation model used is the same as described for Region B.

3.1.3 Region C

The Region C storage cell is modeled in KENO as a square cell that is 9.00 inch in pitch. The open stainless steel box extends inward half of the nominal thickness of 0.135 inch and the inner rack dimension is 8.865 inch. The rack dimensions are seen in Figure 3.1-3.

The three fuel storage cell types used in Region C are as follows:

1. Westinghouse fuel assembly with no poison rodlets inserted into guide tube cells.

2. Westinghouse fuel assembly with a poison rodlets inserted into guide tube cells that occupy a single diagonal direction.

3. Consolidated fuel.

Geometric specifications for Westinghouse fuel type are listed in column four of Table 3.2-1.

Poison RODLETS are defined in KENO input as cylinders of 0.87 inch outside diameter that extend axially to 3 inch above the active fuel bottom. The Westinghouse fuel assembly with a poison rod inserted into three diagonally oriented guide tube cells is shown Figure 3.2-3, that was captured from the KENO plotter output. Two different diagonal orientations were investigated. The four zone axial burnup profile is constructed in KENO as four cylinders stacked axially.

3.1.4 Full Spent Fuel Pool Model

The KENO GIF plot for the Millstone Unit 2 spent fuel pool as modeled by KENO is shown in Figure 3.1-4. All Region C storage cells are loaded with fuel assemblies enriched to 4.0 w/o U-235 and depleted to 45,000 MWD/MT, which is more conservative than the required burnup. Region B was modeled with Siemens fresh fuel assemblies enriched to 4.85 w/o U-235 in 3 of each 4 storage locations. The 4th location in Region B contained Batch B fuel assemblies with an initial enrichment equal to 2.36 w/o U-235 and depleted to 22,300 MWD/MT. The fuel assemblies modeled in Region A cells are Siemens fuel assemblies with an initial enrichment equal to 3.72 w/o U-235 and burnup equal to 0 MWD/MT, which is more conservative than the required 3.64 w/o enrichment at zero burnup. The fuel elevator is loaded with a single fresh Siemens fuel assembly enriched to 4.85 w/o U-235. All intramodule gaps of water were modeled as 0.1 inches thick.

Figure 3.1-1
Sketch of KENO Model for Infinite Array of Region B Cells

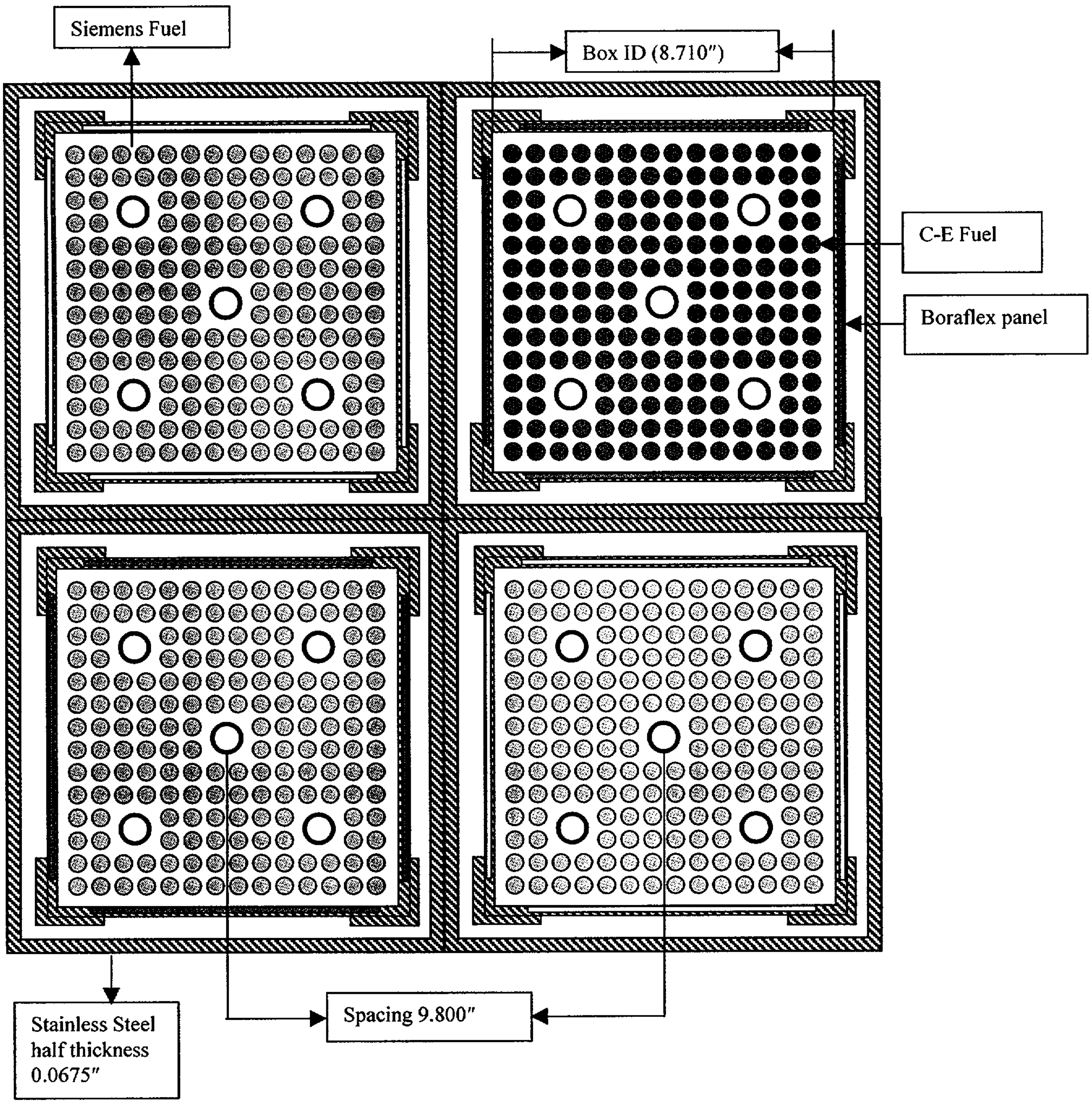


Figure 3.1-2
Sketch of KENO Model for Infinite Array of Region A Cells

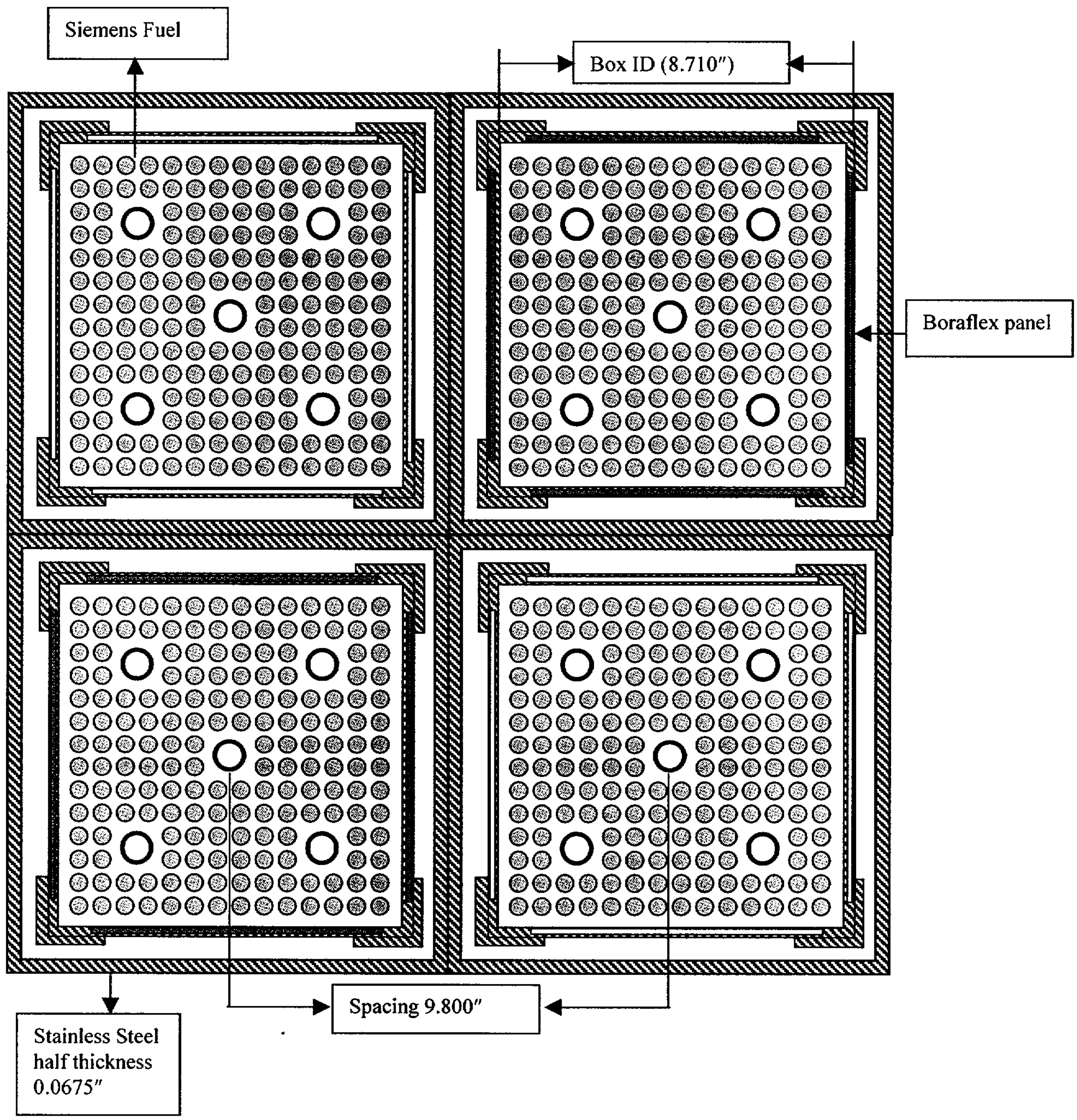


Figure 3.1-3
Sketch of KENO Model for Infinite Array of Region C Cells Without Poison Pins

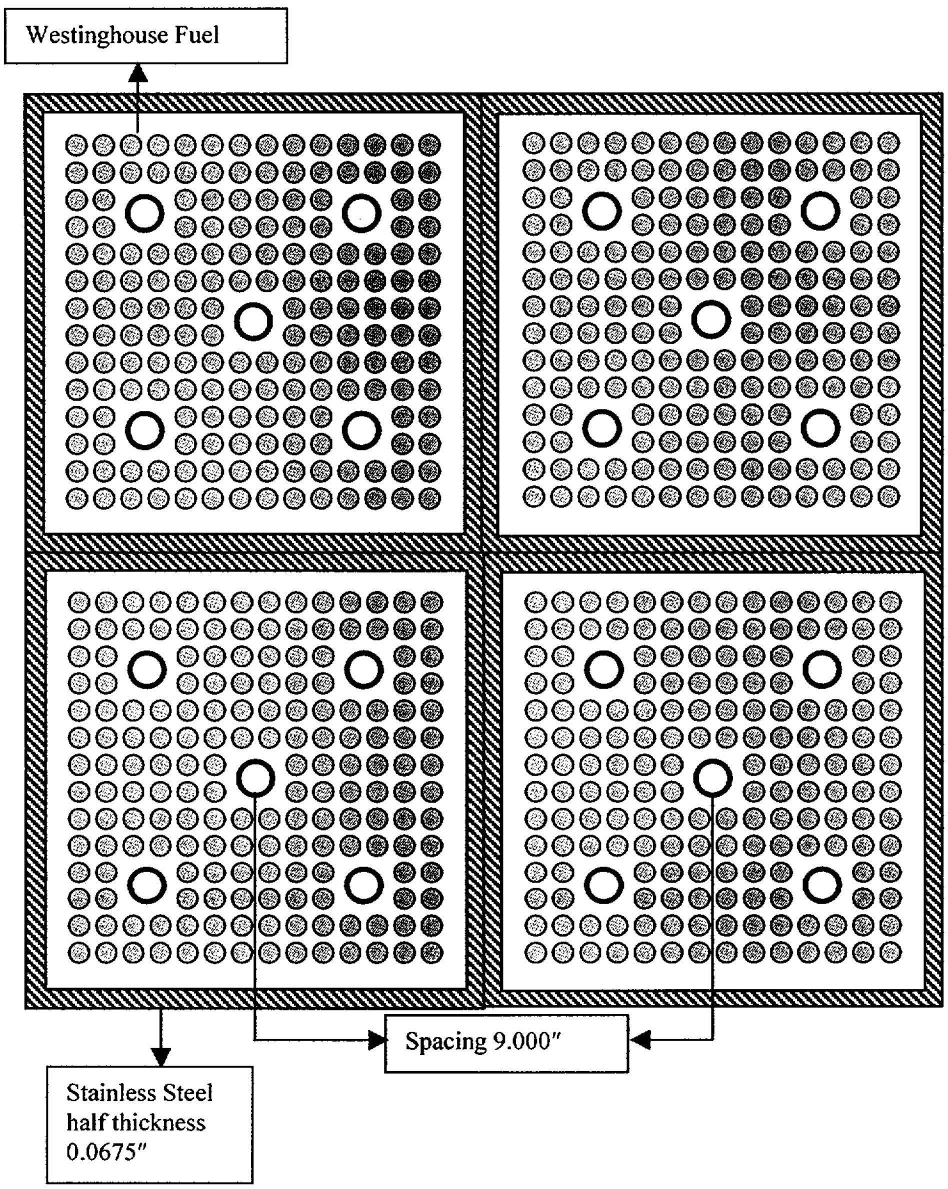
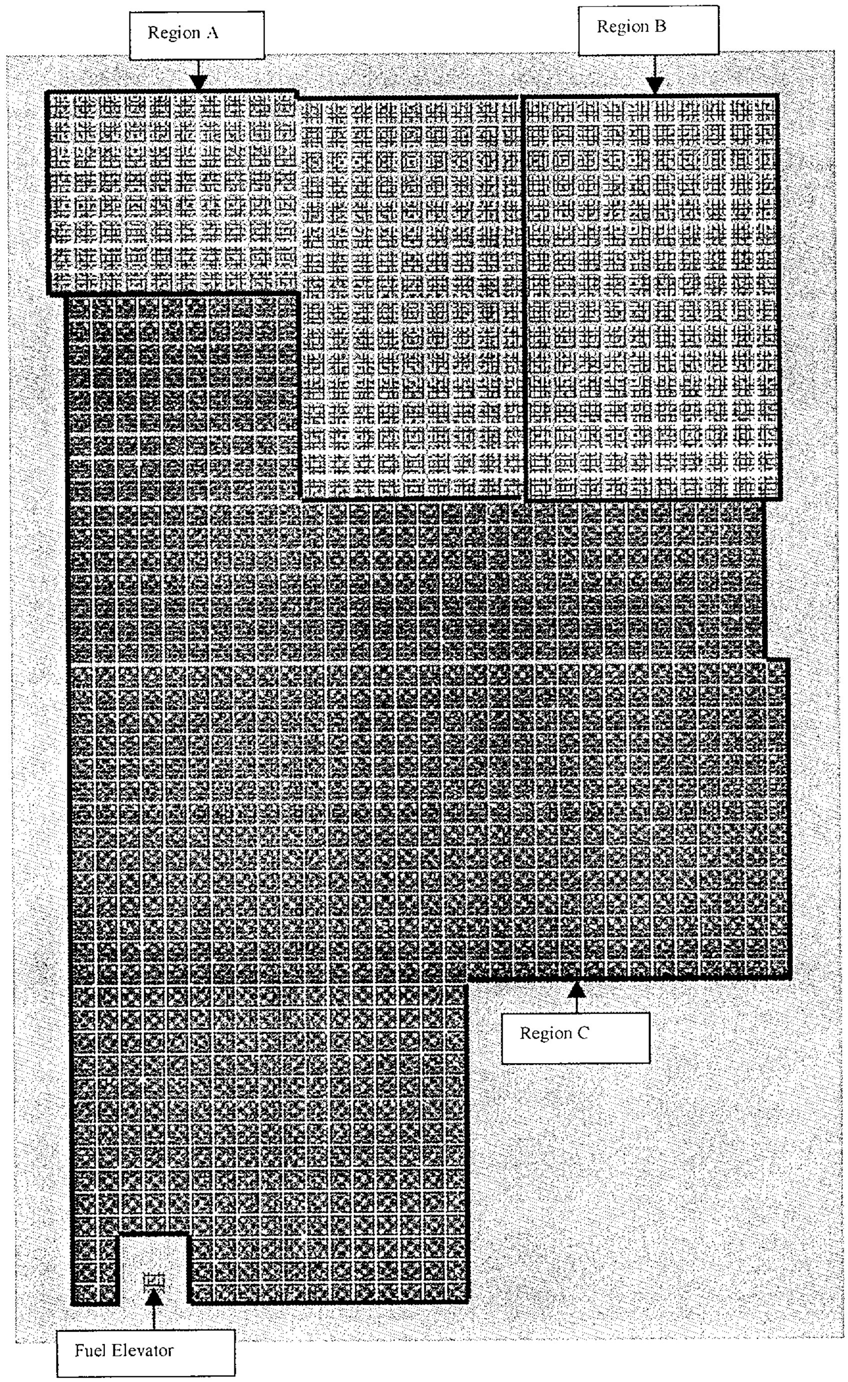


Figure 3.1-4
KENO GIF Plot of the Full Spent Fuel Pool Model



3.2 Design Basis Fuel Assemblies

The Millstone Nuclear Power Station Unit 2 has been in operation for more than 25 years and during that time interval a variety of reload batches containing different fuel assembly designs have been cycled through the reactor. Thus, the criticality safety analysis of the spent fuel pool must take into account possible differences in the reactivity characteristics of the different assembly types. For purposes of this analysis, the different types of fuel assemblies were surveyed so as to define a reference design fuel assembly, for each region, which would assure conservative results for all assembly types.[4]

Table 3.2-1 provides the relevant dimensions of the Siemens, C-E, Westinghouse 14X14 fuel assemblies in the spent fuel pool environment. The 14x14 fuel assembly is shown in Figure 3.2-1.

3.2.1 Design Basis Fuel Assembly For Regions A and B

The Siemens fuel assembly was determined to be the most reactive fuel assembly to represent fresh and low burnup fuel assemblies in Region B and A respectively of the Millstone Unit 2 spent fuel pool. This assembly design was chosen since it is the only assembly design in the spent fuel pool with planar enrichments greater than 4.5 w/o U-235.

3.2.2 Design Basis Fuel Assembly For Region C

The Westinghouse 14x14 fuel assembly design with a fuel density equal to 95.35 % TD was determined to be the most reactive fuel assembly to represent discharged fuel assemblies in Region C of the Millstone Unit 2 spent fuel pool. It is noted that the actual Westinghouse assemblies residing in the spent fuel pool were manufactured with a pellet density approximately equal to 94 % TD. The conservative representation of the pellet density in the Westinghouse assembly design was the key factor in determining the design basis fuel assembly for Region C.

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[4] Note that none of the fuel assembly models assumed the presence of axial blankets since unblanketed fuel produces the limiting k_{eff} for the criticality analysis.

Table 3.2-1
Input Parameters for the Siemens, CE, and Westinghouse 14X14 Fuel Assembly Models

Description	Siemens[5]	CE[6]	Westinghouse[7]
Number of Fuel Rods	176	176	176
Guide Tubes/ Assy.	5	5	5
Rod Pitch, inch	0.580	0.580	0.580
Pellet OD, inch	0.377	0.3795	0.3805
Pellet Density, % TD	95.35 ± 1.5	91.66	95.35
Max. Enrichment, wt%	4.85	2.36	4.85
Active Fuel Length, inch	136.9	136.9	136.9
Clad OD, inch	0.440	0.440	0.440
Clad Thickness, inch	0.028	0.026	0.026
Clad Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Guide Tube OD, inch	1.115	1.115	1.111
Guide Tube ID, inch	1.035	1.035	1.035
Guide Tube Mat.	Zircaloy-4	Zircaloy-4	Zircaloy-4
Poison Rodlet OD, inch	n/a	n/a	0.87 ± 0.015
Poison Rodlet Mat.	n/a	n/a	BSS

[5] Most reactive fuel assembly design for Regions A and B. This is the most conservative of the Siemens fuel designs currently used at Millstone 2.
[6] CE initial core fuel assembly design used in Region B.
[7] Conservative representation with respect to fuel density.

Figure 3.2-1
Geometrical View of 14x14 Standard Fuel Assembly

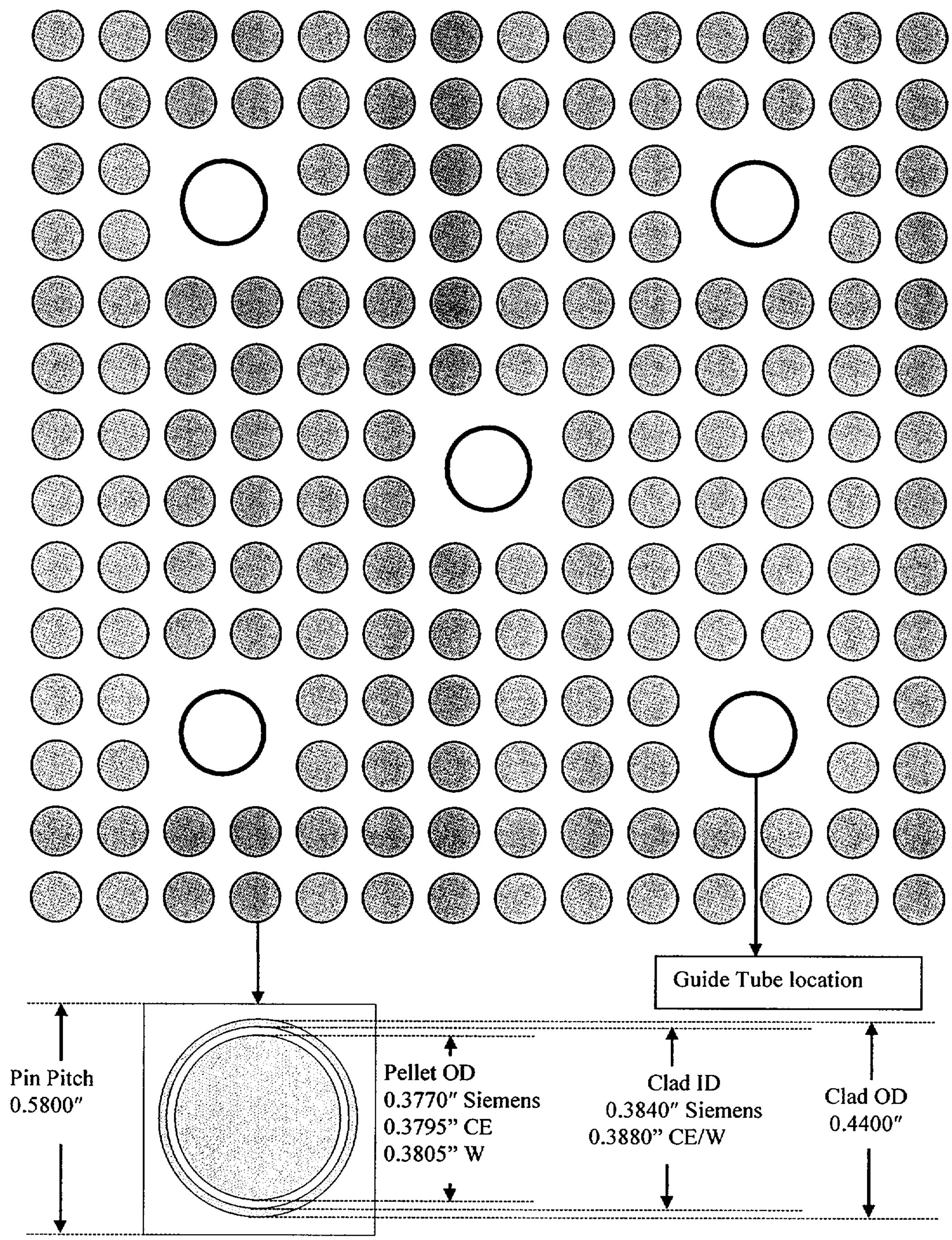
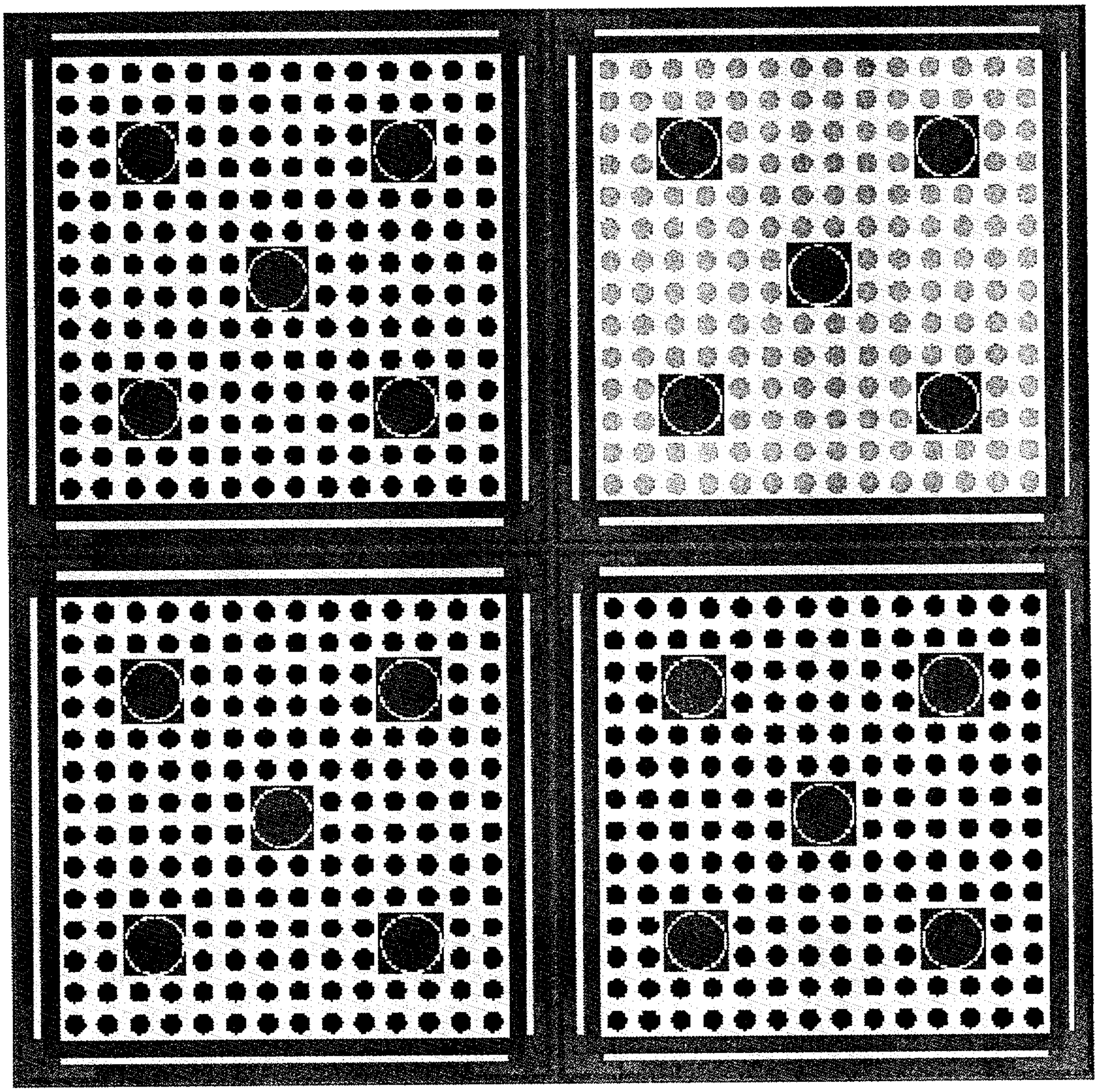


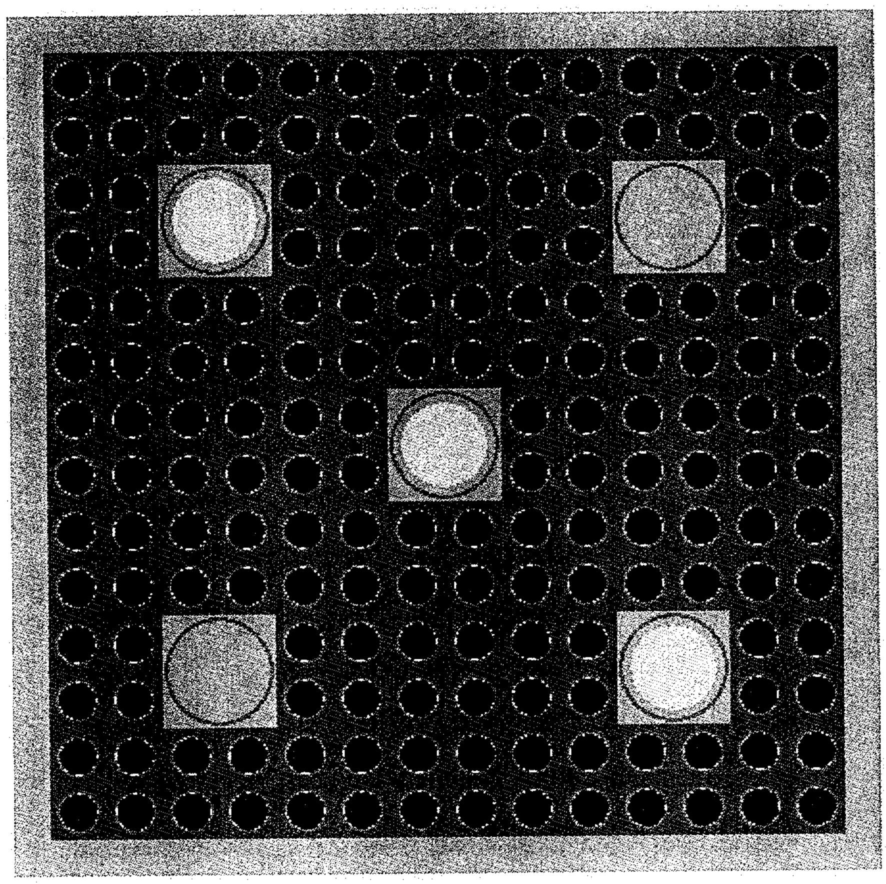
Figure 3.2-2
Region B Storage Cells from KENO Plotter



LEGEND:

- Void
- Siemens (current) fuel
- Zircaloy-4
- Fuel Assembly Water
- Boraflex (0.025 g B10/sc)
- SS304
- Storage Cell Water
- C-E Batch B fuel

Figure 3.2-3
Region C Storage Cells with RODLETS from KENO Plotter



LEGEND:

Westinghouse fuel
Zircaloy-4
Fuel assembly water
Borated SS304 (2 %wt. B-10)
Storage cell water
SS304

Note: Gap between Pellet and Clad appears as the color white on this figure.

3.3 Modeling of Axial Burnup Distributions

A key aspect of the burnup credit methodology employed in this analysis is the inclusion of an explicit axial burnup profile correlated with feed enrichment and discharge burnup of the burned fuel assemblies. This effect is important in the analysis of the spent fuel pool characteristics since the low burnup region at the top and bottom of the fuel assembly will result in a higher reactivity than an assembly having the same average burnup but with less axial variation. In addition these have a discharge burnup well beyond the limit for which the assumption of an uniform axial burnup shape is conservative. Therefore, it is necessary to represent the burnt fuel assembly with a representative axial burnup profile.

For any given spent fuel assembly, the fuel burnup is a continuous function of axial position. However, from a calculational point of view, this function can be discretized in such a manner that the axial "end-effect" is adequately captured. In the methodology used here the fuel assembly is separated by several axial zones with each zone assumed to be uniform in burnup. The size of the top and bottom axial zones are small (typically 6 to 8 inches) so as to capture the steep burnup gradient with axial position while that of the central zone may be larger. In spent fuel pool calculations, a four-zone axial model is found to be conservative (Reference 15, PE&D Report) to represent the spent fuel assembly. Such a four-zone model has three zones with fine mesh spacing (three at the top of the fuel assembly) and the fourth zone is the remainder of the fuel assembly. Figure 3.3-1 provides a pictorial view of the axial zones employed in the four zone axial model.

The individual power fractions of each zone are chosen to yield the same volume averaged burnup when compared to a uniform burnup model. This model is validated due to the fact that the relative contribution of the bottom zones of the fuel assembly to the k_{eff} is negligible. A benchmarking comparison of the assembly k_{eff}, in the spent fuel pool environment, of the four-zone model and a multi-zone (seven-zone) model, performed in Reference 15 provides adequate validation for the use of such a simplified model for 3D representation of burned fuel assemblies.

Input to this analysis is based on the limiting axial burnup profile data provided in the DOE Topical Report, as documented in Reference 14. The burnup profile in the DOE Topical Report is based on a database of 3169 axial-burnup profiles for PWR fuel assemblies compiled by Yankee Atomic. This profile is derived from the burnups calculated by utilities or vendors based on core-follow calculations and in-core measurement data. The axial burnup profile in the DOE report is based on the most limiting axial burnup shape found in the database. The four zone model is constructed based on this limiting axial burnup profile[8].

[8] Note that, although the burnup profiles are representative of fuel without axial blankets, it is appropriate for use in the present analysis since the addition of low enriched blanket regions would produce lower k-eff values at all assembly burnup values.

DIT was used to generate the isotopic concentrations for each segment of the axial profile. Table 3.3-1 lists the fuel and moderator temperatures employed in the spectral calculations for each node in each of the four-zone axial burnup model. These values are based on mid-cycle temperature profiles for the Millstone Unit 2 PWR and are documented in Reference 16. These node dependent moderator and fuel temperature data and power profile data were employed in DIT to deplete the fuel to the desired burnup for each initial enrichment and each axial zone. The values of assembly average burnups versus feed enrichment for which burned fuel assemblies were simulated are tabulated in Table 3.3-2.

A constant soluble boron concentration of 800 ppm was employed in all the DIT calculations in order to obtain a representative neutron spectrum for the isotopic depletion. This value is representative of a cycle average soluble boron concentration in the Millstone Unit 2 core. For the purpose of extracting the number densities, the DIT computer code was executed in two modes. First, a normal depletion was continued in steps of 1000 MWD/MT (with respect to the assembly average case) until the desired burnup was reached. Then a restart is performed at cold, spent fuel pool conditions and the fuel assembly is allowed to decay for 100 hours. At this point of time, the reactivity of the burned fuel assembly is at its highest. The k_{inf} and the isotopic number densities are then extracted for the KENO model development at these assembly conditions.

The DIT computed isotopic concentrations were transferred into the KENO models of the storage cells using a limited set of isotopes. That is, the ^{235}U, ^{238}U, ^{236}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{16}O, and equilibrium ^{149}Sm at shutdown are represented explicitly in the KENO models. All other fission product isotopic number densities are represented by an equivalent ^{10}B concentration; the magnitude of this concentration is determined by matching the DIT k_{eff} value with the KENO k_{eff} computed for an infinite array of in-core assemblies to a one sigma tolerance level.

Appendix A contains a listing of the isotopic number densities employed in the KENO calculations. The format of the listing is compatible with the KENO input description and can directly be used as part of KENO input for material specification. The isotopic number densities are listed for the combination of initial enrichment and burnup listed Table 3.3-2. The listing is for the Westinghouse 14X14 Standard fuel assembly design.

Appendix B contains a listing of the ^{10}B number densities determined by matching the DIT k_{eff} and KENO calculated K_{eff} values. There are a total of 10 tables provided in this Appendix. The ^{10}B number density, the DIT calculated k_{eff} and the KENO calculated k_{eff}, for the four zone axial model (and the average fuel assembly model) are listed in each table. The first three tables contain these values for 3.0 w/o, the next four tables contain the data for 4.0 w/o and the next three tables contain data for 5.0 w/o.

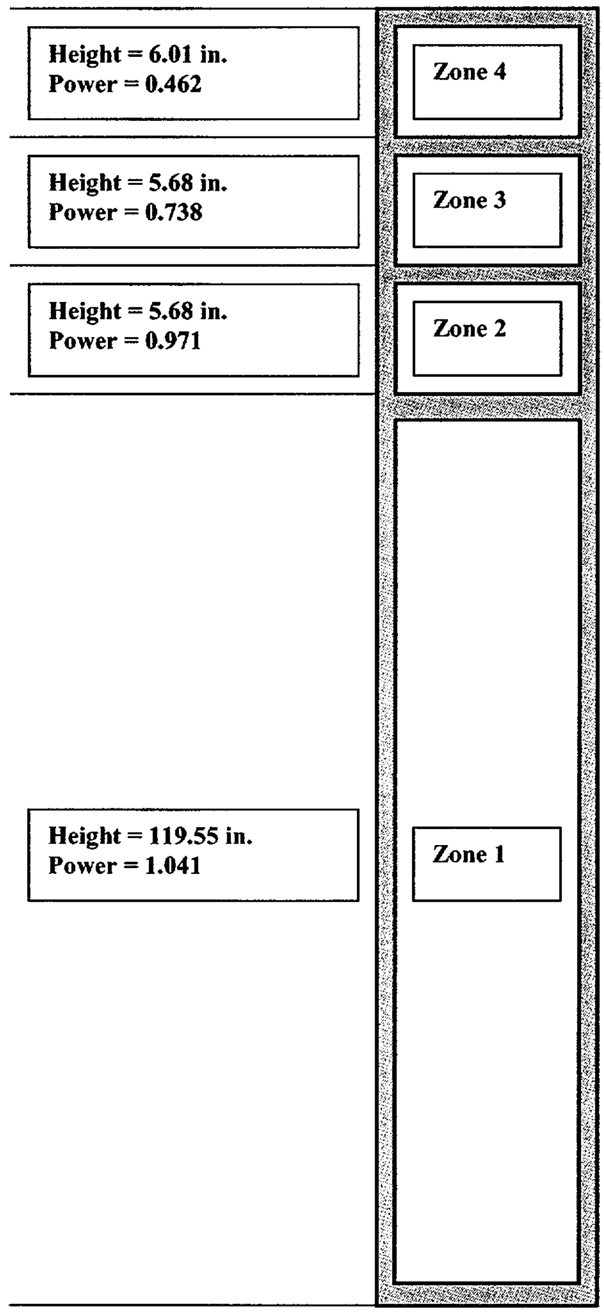
Table 3.3-1
Relative Power, and Fuel and Moderator Temperatures for the Four Zone Model

Zone No.	Height (in.)	Relative Power	Fuel Temperature (°F)	Moderator Temperature (°F)
Average	136.91	1.000	1193.0	574.0
1(bottom)	119.55	1.041	1216.0	571.6
2	5.68	0.971	1129	595.3
3	5.68	0.738	1051	597.0
4(top)	6.01	0.462	919	598.4

Table 3.3-2
Burnup and Initial Enrichments combinations used to determine the Isotopic Number Densities

3 wt%	4 wt%	5 wt%
MWD/MT	MWD/MT	MWD/MT
15,000	25,000	0
25,000	35,000	45,000
35,000	45,000	55,000
	55,000	65,000

Figure 3.3-1
Sketch of Axial Zones Employed in Fuel Assembly



3.4 Tolerance/Uncertainty Evaluation for Spent Fuel Pool Cells

Previous sections described the three region storage racks within the spent fuel storage pool and the KENO model employed to represent infinite arrays of individual cell types and arrays in these storage racks. In addition, the method of modeling the axial profiles of fuel assembly burnup, moderator temperature, and fuel temperature were discussed in so far as their use in reactivity equivalencing fuel assemblies of different burnup histories are concerned.

Using the above input, analytic models were developed to perform the quantitative evaluations necessary to demonstrate that the effective multiplication factor for the spent fuel pool is less than unity with zero boron present in the pool. Applicable biases to be factored into this evaluation are: (1) the methodology bias deduced from the validation analyses of pertinent critical experiments, and (2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range that is less than or equal to 150 °F.

A second allowance is based on a 95/95 confidence level assessment of tolerances and uncertainties; included in the summation of variances are the following:

a) the 95/95 confidence level methods variance,

b) the 95/95 confidence level calculational uncertainty,

c) fuel rod manufacturing tolerance,

d) storage rack fabrication tolerances,

e) tolerance due to positioning the fuel assembly in the storage cell.

Items a) and b) are based on the calculational methods validation analyses of Reference 12. For Item c), the fuel rod manufacturing tolerance for the reference design fuel assembly is assumed to consist of four components: an increase in fuel enrichment from 4.85 to 4.90 wt% ^{235}U and an increase in pellet density from 95.35 to 96.85 %TD; the individual contributions of each change are combined by taking the square root of the sum of the squares of each component. There is no allowance for dishing and chamfer and therefore the pellet density conservatively represents the stack density of the UO_2 pellets in the fuel rod.

For Item d), the following uncertainty components are evaluated. For the Region A and B racks, the stainless steel box ID is decreased from 8.71 to 8.66 inches, the Stainless Steel box thickness is decreased from 0.135 inches to 0.123 inches, the storage rack pitch is decreased from 9.8 to 9.71 inch and the Boraflex width is decreased from 8.063 to 8.000 inch.

In the case of Region C, only two rack tolerances are applicable. The storage rack pitch is decreased from 9.0 to 8.91 inch and the Stainless Steel box thickness is decreased from 0.135 inches to 0.123 inch. The BSS Rodlet outer diameter is decreased from 0.870 inch to 0.855 inch and the weight percent of natural Boron is reduced from 2.0 %wt. B-10 to 1.9 %wt. B-10. An additional bias was also included in the Region C analysis to represent the potential increase in k_{eff} associated with alternate orientation of the BSS Rodlets.

In the case of the tolerance due to positioning of the fuel assembly in the storage cells, all nominal calculations are carried out with fuel assemblies conservatively centered in the storage cells. One case was run to investigate the effect of off-center position of the fuel assembly within the rack cells. This case positioned the assemblies as close as possible in four adjacent storage cells. A negative bias is observed by positioning fresh Westinghouse fuel type assemblies of 95.35 % TD to the center of a 4x4 array of Region C storage cells in KENO. Eccentric positioning has a negative reactivity effect for Regions A, B and C storage cells.

For Region C, a bias for spent fuel pool temperature and BSS RODLET orientation were calculated. The bias for spent fuel pool temperature was evaluated by modeling the maximum spent fuel pool temperature, 150 °F, directly with KENO. The spent fuel pool temperature bias was calculated by subtracting the KENO result for the nominal condition (68 °F), minus the associated uncertainty, from the KENO result for the maximum spent fuel pool temperature plus the associated uncertainty. The BSS RODLET orientation bias was calculated by subtracting the k_{eff} value for the nominal orientation minus the associated uncertainty from the k_{eff} value, plus the associated uncertainty, for the most reactive BSS RODLET orientation.

Table 3.4-1 provides a summary of the KENO cases used in the calculation of biases and uncertainties for the zero soluble boron condition in the infinite array models for Region B and Table 3.4-2 provides similar results for Region A. Table 3.4-3 provides a summary of uncertainties and biases for Region C. The Region C bias and uncertainty is added to the eigenvalue calculated for the zero soluble boron condition in the entire spent fuel pool model.

Table 3.4-1
Keno Calculated K_{eff} values for the Various Physical Tolerance Cases in Region B

Case Description	k-eff	Delta k-eff
Nominal Case	0.97628 ± 0.00033	
Increase in U-235 Enrichment	0.97809 ± 0.00033	0.00247
Increase in Stack Density	0.97871 ± 0.00034	0.00310
Decrease in Boraflex Width	0.97749 ± 0.00033	0.00187
Increase in Cell ID	0.97943 ± 0.00033	0.00381
Decrease in Pitch	0.99017 ± 0.00033	0.01455
Decrease in Rack Thickness	0.97581 ± 0.00033	0.00019
Off-Center Assembly Positioning		NEGATIVE
KENO Uncertainty[9]		0.00664
Statistical Sum of Tolerances		0.01701
Methodology Bias[10]		0.00259
Sum of Tolerances and Biases		0.01960

[9] KENO Uncertainty is the sum of the evaluation of the right hand terms in Equation 3 (see Equation 8 for specific values of the mean calculational variance, and the 95/95 confidence level multiplier) using a maximum k-eff σ of 0.00080.
[10] "Methodology Bias" or the mean calculational methods bias is evaluated to be 0.00259 using Equation 4.

Table 3.4-2
Keno Calculated K_{eff} values for the Various Physical Tolerance Cases in Region A

Case Description	k-eff	Delta k-eff
Nominal Case[11]	0.97409 ± 0.00033	
Increase in Stack Density	0.97602 ± 0.00033	0.00259
Increase in U-235 Enrichment	0.97730 ± 0.00034	0.00388
Decrease in Boraflex Width	0.97460 ± 0.00034	0.00118
Increase in Cell ID	0.98083 ± 0.00033	0.00740
Decrease in Pitch	0.98887 ± 0.00032	0.01543
Increase in Rack Thickness	0.97593 ± 0.00033	0.00250
Off-Center Assembly Positioning		NEGATIVE
KENO Uncertainty[12]		0.00664
Statistical Sum of Tolerances		0.01914
Methodology Bias[13]		0.00259
Sum of Tolerances and Biases		0.02173

[11] Note, the uncertainties for Region A were derived based upon an initial enrichment equal to 3.72 w/o U-235. However the initial enrichment was reduced to 3.64 w/0 U-235 to achieve a target k-eff value approximately equal to 0.970.
[12] KENO Uncertainty is the sum of the evaluation of the right hand terms in Equation 3 (see Equation 8 for specific values of the mean calculational variance, and the 95/95 confidence level multiplier) using a maximum k-eff σ of 0.00080.
[13] "Methodology Bias" or the mean calculational methods bias is evaluated to be 0.00259 using Equation 4.

Table 3.4-3
Keno Calculated K_{eff} values for the Various Physical Tolerance Cases in Region C

Case Description	k-eff	Delta k-eff
Fuel Nominal Case	1.38929 ± 0.00057	
Increase in U-235 Enrichment	1.39025 ± 0.00058	0.00211
Increase in Stack Density	1.38938 ± 0.00054	0.00120
Off-Center Assembly Positioning	1.35910 ± 0.00060	NEGATIVE
Rack Nominal Case	0.97420 ± 0.00054	
Decrease in Pitch	0.98351 ± 0.00053	0.01038
Decrease in Rack Thickness	0.97787 ± 0.00055	0.00476
BSS Rodlet Nominal Case	0.89914 ± 0.00054	
Reduced BSS Rodlet Diameter	0.89914 ± 0.00054	0.00108
Reduced BSS Boron	0.90125 ± 0.00053	0.00318
KENO Uncertainty[14]		0.00664
Statistical Sum of Tolerances		0.01384
Temperature Bias (68°F to 150°F)	0.97955 ± 0.00052	0.00641
Staggered BSS Rodlet Bias	0.90021 ± 0.00054	0.00215
Methodology Bias[15]		0.00259
Sum of Tolerances and Biases		0.02499

[14] KENO Uncertainty is the sum of the evaluation of the right hand terms in Equation 3 (see Equation 8 for specific values of the mean calculational variance, and the 95/95 confidence level multiplier) using a maximum k-eff σ of 0.00080.
[15] "Methodology Bias" or the mean calculational methods bias is evaluated to be 0.00259 using Equation 4.

3.5 No Soluble Boron 95/95 K_{eff} Calculation Results

The fuel assemblies modeled in Regions A, B and C were uniformly enriched in the radial direction. Typically, the fuel pins around the large water holes of this fuel assembly design are enriched to lower values to flatten the power distribution of the fuel assembly. This means that individual fuel pins located away from the large water holes may have enrichments greater than the radially averaged enrichment value. A distributed fuel pin enrichment, designed to flatten the assembly power distribution, produces a less reactive assembly design compared to an assembly with a uniform enrichment distribution. For this reason, the analysis results for Region A, B and C will be quoted as a "radially-averaged fresh fuel enrichment". In summary, the use of a radially averaged fuel enrichment is conservative relative to the distributed enrichments actually used in designs of this fuel assembly type.

3.5.1 Region B Cells

As described in section 3.2.1 Region B consists of fresh Siemens fuel assemblies in 3-out-of-4 available adjacent fuel storage locations with the remaining location containing a burnt CE fuel assembly of 2.36 % U-235 by weight and a burnup of 22,300 MWD/MT. The eigenvalue for an infinite array of Region B storage cells is shown Table 3.4-1 (and also in Table 3.5-1. Calculations were run at 68°F, with maximum water density consistent with 39°F, to maximize k_{eff}. The calculated KENO multiplication factor for the Region B infinite array is .97628 ± 0.00033, without biases and uncertainties, with no soluble boron. The biases and uncertainties, also from Table 3.4-1, for Region B is computed to be 0.01960 delta K-effective units. Therefore, the final 95/95 K-effective value at zero soluble for an infinite Region B model is 0.99588. This value is below the design basis limit equal to unity at zero soluble boron.

3.5.2 Region A Cells

Region A is analyzed for an infinite array of 4-out-of-4 fuel storage of 3.64 % U^{235} by weight with no burnup using the Siemens fuel design. Table 3.5-2 lists k_{eff} values for this fresh fuel and 5.00 % U^{235} by weight initial enrichment fuel at burnup points of up to 10,000 MWD/MT. Calculations were run at 68°F, with maximum water density consistent with 39°F, to maximize k_{eff}.

From Table 3.4-2 the sum of the biases and uncertainties for Region A is 0.02173 delta-k_{eff}. The required assembly burnups as a function of initial enrichment are shown in Table 3.5-5, and these points are fitted to a linear curve. This linear curve and coefficients are shown in Figure 4.2-1, and it was used to determine the burnup required to maintain a k_{eff} less than 0.970 for all burned fuel assemblies as a function of initial enrichment. Thus the final k_{eff} for Region A for an infinite array is 0.970 plus the uncertainty of 0.02173 from Table 3.4-2. This value of 0.99173 is less than the target k_{eff} of 0.995, and less than the design basis limit of 1.0 with zero soluble boron.

3.5.2.1 Region A Cells Without Boraflex Poison Box and Fuel Assembly

It is possible, for the purpose of in-service surveillance testing of the boraflex, that poison boxes may need to be removed from Region A and destructively tested. If replacement poison boxes are not available, then it will be shown that removal of the poison box is acceptable provided no fuel is placed in the cells with the removed poison boxes.

To investigate the reactivity effect of removing a Boraflex poison box and a fuel assembly from each of two storage locations in Region A, an 8 by 9 array of storage cells containing the most reactive fuel type was modeled in KENO. The 8 by 9 array contains two water storage cells with the poison boxes removed, such that only water is left inside the SS304 rack. The two "empty" cells are positioned near the center of the 8 by 9 array as shown in Figure 3.5-1. The KENO k_{eff} value for this case produced a reactivity decrease of 0.00472 delta-k_{eff} units from the nominal case k_{eff} for an 8 by 9 module containing all Siemens fuel assemblies and Boraflex poison boxes. The conclusion is that removal of poison boxes is acceptable, providing that no fuel is contained in the cells within the removed poison boxes. Also, if more than 1 location has a removed poison box, there should be at least 3 intervening storage cells separating the locations with the removed poison boxes.

3.5.3 Region C Cells

As described in Section 3.3, the spent fuel storage rack analysis model employs a four axial zone representation of spent/burned fuel assemblies in the storage racks in an evaluation of burnup credit for a given spent fuel assembly. For the region C storage cells, k_{eff} was evaluated for an infinite array of storage cells over a range of feed enrichment values up to 5 wt.% ^{235}U and fuel assembly average burnups up to 65,000 MWD/MT. Calculations were performed at 68°F, and a bias was added to Table 3.4-3 to account for the fact that maximum reactivity for Region C was at 150°F. The storage arrangement that was analyzed, based on Figure 3.1-3 consisted of a four-out-of-four fuel arrangement (four burned fuel assemblies in a 2x2 array). These data are then employed to determine the burnup limits versus initial feed enrichment (for the burned fuel assemblies) for a given target k_{eff} value at zero soluble boron. A parallel set of calculations are conducted for a diagonal arrangement of borated stainless steel rodlets that are inserted into three of five guide tube locations. The target value of k_{eff} of 0.970 was selected to be less than unity by an amount sufficient to cover the expected magnitude of analytical biases and uncertainties in these analyses, i.e., (0.02499 from Table 3.4-3).

Table 3.5-3 lists the KENO k_{eff} values computed with the four axial zone model for the Region C storage cells versus feed enrichment, and fuel assembly average burnup, with no poison rodlets. Table 3.5-4 lists the KENO k_{eff} values computed with four axial zone model for the Region C storage cells with borated stainless steel rodlets inserted versus feed enrichment, and fuel assembly average burnup. The burnup limits determined based on a linear interpolation of these results are provided in Table 3.5-6.

The assembly burnups required to maintain k_{eff} less than 0.970 as a function of initial enrichment are fitted to a third order polynomial, and are shown in Figure 4.2-2 and Figure 4.2-3. Thus the final k_{eff} for region C is 0.970 plus the uncertainty of 0.02499 from Table 3.4-3. This value of 0.99499 is less than the target k_{eff} of 0.995, and less than the design basis limit of 1.0 with zero soluble boron.

3.5.4 Entire Spent Fuel Pool

A KENO model for the entire spent fuel pool at the zero boron condition was constructed for this analysis. The entire spent fuel pool model is shown in Figure 3.1-2. The calculated KENO multiplication factor for the spent fuel pool using this model is 0.97086 ± 0.00052, without biases and uncertainties, with no soluble boron. As can be seen from Table 3.4-1, Table 3.4-2, and Table 3.4-3, the Region C uncertainties are slightly higher than the Region A and B uncertainties. Therefore only the Region C uncertainty will be used for the KENO result determined for the entire spent fuel pool. This value from Table 3.4-3 is 0.02499. Therefore the final k_{eff} value with no soluble boron is 0.99585 which is less than the design basis limit of 1.0 with zero soluble boron.

Table 3.5-1
KENO K_{eff} Values versus Feed Enrichment, and Assembly Average Burnup for the Region B Storage Cells with No Soluble Boron

No.	Burned Fuel Assembly Description	K_{eff} value from KENO
1	Siemens Fuel in 3-out-of-4 at 4.85 w/o ^{235}U and Burnup = 0 MWD/MT. CE Fuel in 1-out-of 4 at Enrichment = 2.36 w/o ^{235}U and Burnup = 22,300 MWD/MT	0.97628 +/-0.00033

Table 3.5-2
KENO K_{eff} Values versus Feed Enrichment, and Assembly Average Burnup for the Region A Storage Cells with No Soluble Boron

No.	Siemens Fuel Assembly Description	K_{eff} value from KENO
1	Enrichment = 3.64 w/o ^{235}U Burnup = 0 MWD/MT	0.96994 ± 0.00075
2	Enrichment = 5.00 w/o ^{235}U Burnup = 6,000 MWD/MT	0.98261 ± 0.00074
3	Enrichment = 5.00 w/o ^{235}U Burnup = 8,000 MWD/MT	0.96752 ± 0.00070
4	Enrichment = 5.00 w/o ^{235}U Burnup = 10,000 MWD/MT	0.95452 ± 0.00068

Table 3.5-3
KENO K_{eff} Values versus Feed Enrichment, and Assembly Average Burnup for the Region C Storage Cells, with No Poison Rodlets, with No Soluble Boron

No.	Westinghouse Fuel Assembly Description	K_{eff} value from KENO
1	Enrichment = 3.00 w/o ^{235}U Burnup = 15,000 MWD/MT	1.10083 ± 0.00049
2	Enrichment = 3.00 w/o ^{235}U Burnup = 25,000 MWD/MT	1.01522 ± 0.00049
3	Enrichment = 3.00 w/o ^{235}U Burnup = 35,000 MWD/MT	0.94756 ± 0.00049
4	Enrichment = 4.00 w/o ^{235}U Burnup = 35,000 MWD/MT	1.03267 ± 0.00052
5	Enrichment = 4.00 w/o ^{235}U Burnup = 45,000 MWD/MT	0.97582 ± 0.00054
6	Enrichment = 4.00 w/o ^{235}U Burnup = 55,000 MWD/MT	0.92550 ± 0.00053
7	Enrichment = 5.00 w/o ^{235}U Burnup = 45,000 MWD/MT	1.04601 ± 0.00052
8	Enrichment = 5.00 w/o ^{235}U Burnup = 55,000 MWD/MT	0.99684 ± 0.00057
9	Enrichment = 5.00 w/o ^{235}U Burnup = 65,000 MWD/MT	0.94996 ± 0.00059

Table 3.5-4
KENO K_{eff} Values versus Feed Enrichment, and Assembly Average Burnup for the Region C Storage Cells with No Soluble Boron and BSS Rodlets Installed

No.	Westinghouse Fuel Assembly Description	K_{eff} value from KENO
1	Enrichment = 3.00 w/o ^{235}U Burnup = 15,000 MWD/MT	1.00850 ± 0.00054
2	Enrichment = 3.00 w/o ^{235}U Burnup = 25,000 MWD/MT	0.92882 ± 0.00051
3	Enrichment = 3.00 w/o ^{235}U Burnup = 35,000 MWD/MT	0.86643 ± 0.00052
4	Enrichment = 4.00 w/o ^{235}U Burnup = 25,000 MWD/MT	1.01773 ± 0.00055
5	Enrichment = 4.00 w/o ^{235}U Burnup = 35,000 MWD/MT	0.95287 ± 0.00052
6	Enrichment = 4.00 w/o ^{235}U Burnup = 45,000 MWD/MT	0.89884 ± 0.00054
7	Enrichment = 4.00 w/o ^{235}U Burnup = 55,000 MWD/MT	0.85204 ± 0.00053
8	Enrichment = 5.00 w/o ^{235}U Burnup = 45,000 MWD/MT	0.96757 ± 0.00057
9	Enrichment = 5.00 w/o ^{235}U Burnup = 55,000 MWD/MT	0.92164 ± 0.00057
10	Enrichment = 5.00 w/o ^{235}U Burnup = 65,000 MWD/MT	0.87715 ± 0.00056

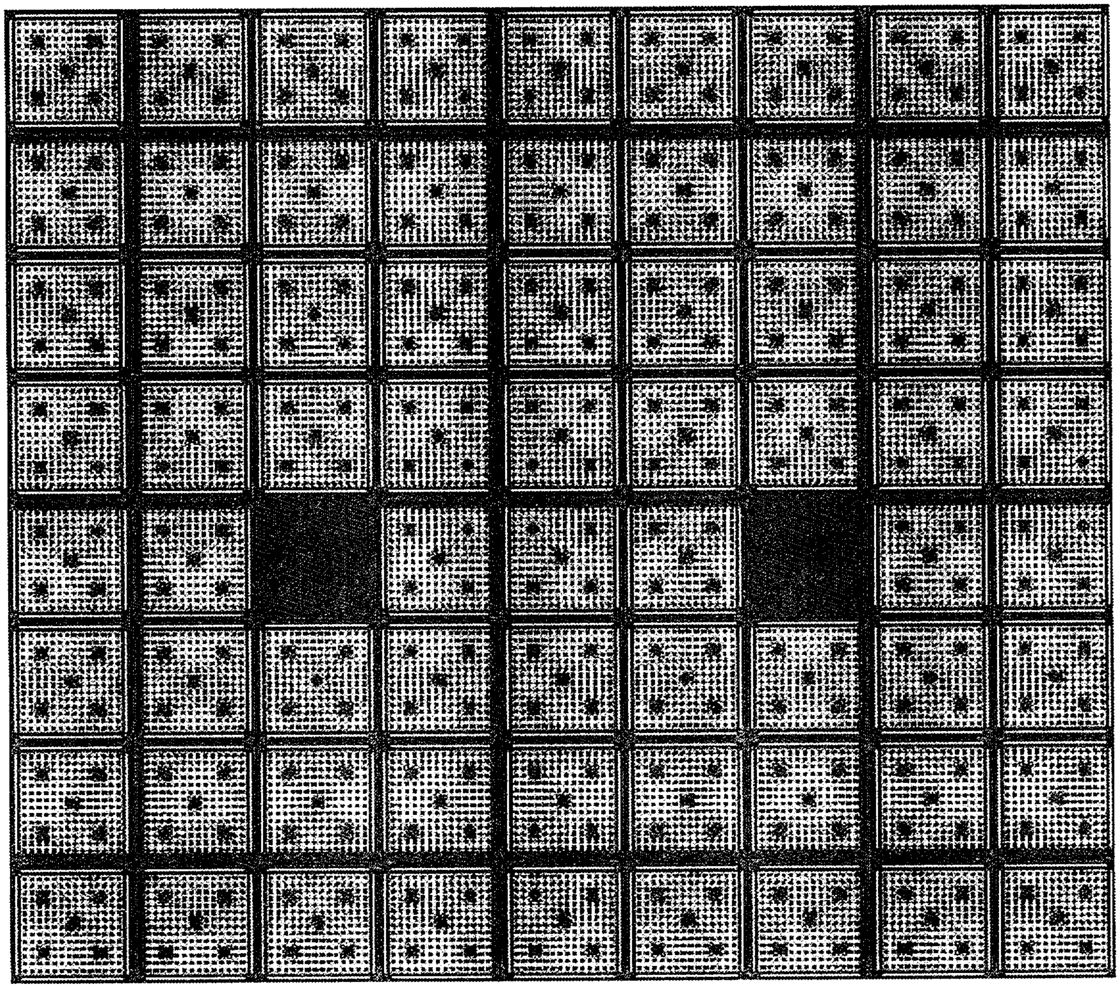
Table 3.5-5
Required Fuel Assembly Burnup (MWD/MT) versus Initial Enrichment for the Region A Storage Cells

Enrichment	Burnup (MWD/MT)
3.64	0
5.00	7,683

Table 3.5-6
Required Fuel Assembly Burnup (MWD/MT) versus Initial Enrichment for the Region C Storage Cells

Enrichment [wt. % U-235]	**Burnup Required with no RODLETS** [MWD/MT]	**Burnup Required with RODLETS** [MWD/MT]
1.1955	0	N/A
1.6250	N/A	0
3	31,490	19,445
4	46,093	32,129
5	60,667	44,479

Figure 3.5-1
KENO GIF Plot of Region A 8 by 9 Module Showing 2 Boraflex Poison Boxes and Siemens Fuel Assemblies Removed



LEGEND:
- Void
- Siemens (current) fuel type
- Zircaloy-4
- Fuel Assembly Water
- Boraflex (0.025 g B-10 /sc)
- SS304
- Storage Cell Water

3.6 Soluble Boron K_{eff} Calculation Results

The NRC Safety Evaluation Report (SER) for WCAP-14416-P is given in Reference 3; Page 9 of the enclosure to Reference 3 defines the soluble boron requirement as follows. The total soluble boron credit requirement is defined as the sum of three quantities:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA} \qquad \textbf{Equation 1}$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm),

$SBC_{95/95}$ = soluble boron requirement for 95/95 $k_{eff} \leq 0.95$ (ppm),

SBC_{RE} = soluble boron required for reactivity equivalencing methodologies (ppm),

SBC_{PA} = soluble boron required for $k_{eff} \leq 0.95$ under accident conditions (ppm).

Each of these terms will be discussed in the following subsections.

3.6.1 Soluble Boron Determination to Maintain K_{eff} Less Than 0.95

Table 3.6-1 contains KENO calculated multiplication factors for the entire Millstone Unit 2 spent fuel pool at 0, 200, 400, 600, 800, 1000, 1200 and 1400 ppm of soluble boron. This data was generated for a conservative representation of discharged fuel assemblies in Region C of the spent fuel pool. From Figure 4.2-2, it can seen that the burnup required to store discharged fuel assemblies with an initial enrichment equal to 5.0 w/ U-235 is approximately 60,667 MWD/MT. In order to conservatively calculate the soluble boron worths, the discharged fuel assemblies were represented with an initial enrichment equal to 5.0 w/o U-235 and a burnup equal to 65,000 MWD/MT.

The last column in Table 3.6-1 is labeled "Delta K_{eff}." delta k_{eff} is equal to the k_{eff} (plus the one sigma value) of the case with soluble boron minus k_{eff} (minus the one sigma value) of the unborated spent fuel pool. Note that this definition of Delta K-effective will produce a slightly conservative soluble boron worth. The reference k_{eff} value of the unborated spent fuel pool (and the associated one sigma value) is given in this table at zero ppm.

The soluble boron worth in Table 3.6-1 will be employed to determine the soluble boron concentration necessary to maintain k_{eff} less than 0.95 (including biases and uncertainties) and to compensate for the reactivity equivalencing methodologies which could increase the multiplication factor of the spent fuel pool.

The amount of soluble boron required to maintain k_{eff} less than 0.95 including biases and uncertainties is determined based on the results from Table 3.6-1. The soluble boron concentration (ppm) required to reduce the k_{eff} of the entire spent fuel pool by 0.050 delta k_{eff} units is conservatively determined by graphical interpolation of the data in Table 3.6-1 to be equal to 390 ppm.

3.6.2 Soluble Boron Determination for Reactivity Uncertainties

The soluble boron credit (ppm) required for reactivity uncertainties was determined by converting the uncertainty in fuel assembly reactivity and the uncertainty in absolute fuel assembly burnup values to a soluble boron concentration (ppm) necessary to compensate for these two uncertainties. The first term, uncertainty in fuel assembly reactivity, is calculated by employing a depletion reactivity uncertainty equal to 0.005 delta k_{eff} units per 30,000 MWD/T of assembly burnup (obtained from Reference 3) and multiplying by the maximum amount of assembly burnup credited in a Region analysis. The highest assembly burnup credited is 58,422 MWD/T; this value is employed for Region C cells at an initial fuel assembly enrichment equal to 4.85 w/o ^{235}U. Therefore, the uncertainty in fuel assembly reactivity is equal to 0.00974 delta k_{eff} units.

The uncertainty in absolute fuel assembly burnup values is conservatively calculated as 5% of the maximum fuel assembly burnup credited in a Region analysis. The maximum fuel assembly burnup credited in this analysis is 58,422 MWD/T. Such a fuel assembly is used in Region C cells at an initial fuel enrichment of 4.85 wt.% ^{235}U. The uncertainty in this burnup value is determined to be 2,921 MWD/T. The reactivity associated with a delta-burnup of 2,921 MWD/T at 55,000 MWD/T for a Region C cell was calculated to be 0.01370 delta k_{eff} units.

The total of these two reactivity effects is equal to 0.02344 (0.01370 + 0.00974) delta k_{eff} units. By graphical interpolation of the data in Table 3.6-1 the soluble boron concentration (ppm) necessary to compensate for this reactivity is equal to 179 ppm.

3.6.3 Soluble Boron Determination to Mitigate Accidents

The soluble boron concentration (ppm) required to maintain k_{eff} less than or equal to 0.95 under accident conditions is determined by first surveying all possible events which increase the k_{eff} value of the spent fuel pool. The accident event which produced the largest increase in spent fuel pool k_{eff} value is employed to determine the required soluble boron concentration necessary to mitigate this and all less severe accident events. The list of accident cases considered include:

- Misloaded fresh fuel assembly into either Region A, B or C
- Misloaded fresh fuel assembly just outside the storage racks.
- Misloaded fresh fuel assembly near the fuel elevator which contains a fresh fuel assembly
- Heavy load accident in Regions A, B, and C
- Dropped fresh fuel assembly on top of the storage racks.
- Seismic Event which would reduce the intramodule gaps
- Spent fuel pool temperature greater than 150 degrees Fahrenheit.

Several fuel mishandling events were simulated with KENO to assess the possible increase in the k_{eff} value of the Millstone Unit 2 spent fuel pool. The fuel mishandling events all assumed that a fresh Siemens fuel assembly enriched to 4.85 w/o U-235 (and no burnable poisons) was misloaded into the described area of the spent fuel pool. These cases were simulated with the KENO model for the entire spent fuel pool ;

- Fresh fuel assembly misloaded into a Region A storage cell without a Boraflex poison box.
- Fresh fuel assembly misloaded into a Region B storage cell without a Boraflex poison box. Note that the fresh fuel assembly was misloaded into a location intended for a Batch B fuel assembly. This is the most reactive location that the misloaded fresh fuel assembly could occupy in Region B.
- Fresh fuel assembly misloaded into a Region C storage cell.
- Fresh fuel assembly misloaded between a fresh fuel assembly (4.85 w/o U-235) in the fuel elevator and Region C. Note that this misloaded assembly was placed in the most reactive position around the fuel assembly in the elevator.

- Fresh fuel assembly placed just outside the storage racks and adjacent to Regions C and A.

Table 3.6-2 contains the KENO calculated multiplication factors for these mishandling events. It can be seen from Table 3.6-2 that the fresh fuel assembly misloaded into the most reactive position around the fresh fuel assembly in the fuel elevator produced the most reactive fuel mishandling event.

The heavy load accident evaluated for Regions A and B were simulated in KENO by collapsing the poison box onto the outside envelope of the intact fuel assembly and collapsing the stainless steel rack onto the collapsed poison box. The thickness of the poison box and the stainless steel rack were maintained in the collapsing process. This model for the Region A and B storage cells minimizes the center to center spacing between adjacent storage cells.

In addition to these mishandling events it is possible to drop a fresh fuel assembly on top of the spent fuel pool storage racks. In this case the physical separation between the fuel assemblies in the spent fuel pool storage racks and the assembly lying on top of the racks is sufficient to neutronically decouple the accident. In other words, dropping the fresh fuel assembly on top of the storage racks will not produce a positive reactivity increase.

The nominal gap between modules is about 2 to 3 inch wide. A seismic event could reduce the nominal intramodule gaps. The KENO model for the entire spent fuel pool was constructed with extremely small intramodule gaps, 0.1 inches Therefore, the KENO model for the entire spent fuel pool employs intramodule gaps which will not be affected by the design basis seismic event. Further, the infinite lattice reactivity results for Region A, B and C do not credit any gap between racks, and these infinite lattice results meet the required reactivity criteria.

The last accident considered is the increased spent fuel pool temperature above 150 degrees Fahrenheit. (The nominal spent fuel pool temperature range is less than or equal to 150 degrees Fahrenheit.). Since Region C contains a slightly positive moderator temperature coefficient (at least in the temperature range up to 150 degrees Fahrenheit), temperatures up to the saturation temperature were considered. The saturation temperature at the bottom of the spent fuel pool (where the pressure is highest, approximately 2 atmospheres) is equal to 248 degrees Fahrenheit. A KENO model for an infinite array of Region C storage cells loaded with depleted fuel assemblies (4.0 w/o U-235 and 45,000 MWD/MT) was developed to address this reactivity increase. The KENO calculated K-effective value for this model is equal to 0.97773 +/- 0.00050. The KENO calculated multiplication factor for exactly the same infinite array at 150 degrees Fahrenheit (from Table 3.6-2) is equal to 0.97955 +/- 0.00050. Thus there is no additional reactivity increase due to the spent fuel pool temperature increase from 150 to 248 degrees Fahrenheit. Temperatures in excess of 248 degrees Fahrenheit will produce voided conditions in the spent fuel pool water. This voiding effect will significantly reduce the multiplication factor of the spent fuel pool.

Table 3.6-3 lists the KENO calculated multiplication factor for the more limiting accidents at a specified soluble boron concentration. By inspection of this data it is clear that the heavy load accident case in Region B required the most soluble boron to mitigate the event. The heavy load accident in Region B requires 790 ppm in order to reduce the multiplication factor back to the reference case (nominal conditions).

3.6.4 Summary of Soluble Boron Requirements

Soluble boron in the spent fuel pool coolant is used in this criticality safety analysis to offset the reactivity allowances for calculational uncertainties in modeling, storage rack fabrication tolerances, fuel assembly design tolerances, and postulated accidents. The total soluble boron requirement, SBC_{TOTAL}, is defined by the following equation.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA} \qquad \textbf{Equation 1}$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm),

$SBC_{95/95}$ = soluble boron requirement for 95/95 $k_{eff} \leq 0.95$ (ppm),

SBC_{RE} = soluble boron required for reactivity uncertainties (ppm),

SBC_{PA} = soluble boron required for $k_{eff} \leq 0.95$ under accident conditions (ppm).

The magnitude of the above components is:

$SBC_{95/95}$ = 390 ppm

SBC_{RE} = 179 ppm

SBC_{PA} = 790 ppm

SBC_{TOTAL} = **1358 ppm**

Therefore, a total of 1387 ppm of soluble boron is required to maintain k_{eff} less than 0.95 (including all biases and uncertainties) assuming the most limiting single accident. Note that these soluble boron concentrations assumes an atomic fraction for ^{10}B equal to 0.199. For a ^{10}B isotopic fraction equal to 0.197, the soluble boron concentrations, required to maintain the same concentration of ^{10}B atoms, would be equal to :

$$SBC_{95/95} = 394 \text{ ppm}$$

$$SBC_{RE} = 180 \text{ ppm}$$

$$SBC_{PA} = 798 \text{ ppm}$$

$$SBC_{TOTAL} = \mathbf{1372\ ppm}$$

Thus a recommended soluble boron level of **1400 ppm** is sufficient to accommodate all the design requirements. Verification of this is demonstrated by the k_{eff} value, without biases and uncertainties, of 0.91135 at 1400 ppm under the limiting accident condition. With biases and uncertainties the k_{eff} value is less the 0.95 criteria.

Table 3.6-1
K_{eff} as a Function of Soluble Boron Level

Cell Type	**Description**	
Region B	Fresh Fuel Assembly = 4.85 w/o ^{235}U in 3 out of 4 storage locations Batch B Combustion Engineering Fuel Assembly in the 4th location (2.36 w/o U-235 and 22,300 MWD/MT)	
Region A	Fresh Fuel Assembly = 3.72 w/o ^{235}U[16]	
Region C	Burned Fuel Assembly = 5.00 w/o ^{235}U, 65,000 MWD/MT[17]	
Soluble Boron Concentration, ppm	**K_{eff}**	**Delta K_{eff}**
0	0.97016 ± 0.00052	0
200	0.94513 ± 0.00050	-0.02401
400	0.91759 ± 0.00050	-0.05155
600	0.89410 ± 0.00050	-0.07504
800	0.87387 ± 0.00049	-0.09528
1000	0.85416 ± 0.00049	-0.11499
1200	0.83327 ± 0.00048	-0.13589
1400	0.81740 ± 0.00047	-0.15177
Second order fit: = $14696x^2 + 6871.9x + 9.4402$		

[16] The enrichment value of 3.72 w/o ^{235}U is conservatively greater than the required enrichment value of 3.64 w/o ^{235}U at zero burnup.
[17] The burnup value of 65,000 MWD/MT at 5 w/o ^{235}U is conservatively greater than the limiting burnup value of 60,667 MWD/MT at 5 w/o ^{235}U and was chosen to minimize soluble boron worth.

Table 3.6-2
K_{eff} for Accident Events

(Reference K_{eff} =0.97086 ± 0.00052)

Description of Accident	K_{eff}	Delta K_{eff}
Fresh Fuel assembly misloaded in to a Region A storage cell that does not have a Boraflex poison box	0.99537 ± 0.00053	0.02556
Fresh Fuel assembly misloaded in to a Region B storage cell	1.01943 ± 0.00052	0.04961
Fresh Fuel assembly misloaded in to a Region C storage cell	1.02841 ± 0.00039	0.05846
Fresh fuel assembly misloaded between fresh fuel assembly in elevator and Region C	1.05990 ± 0.00050	0.09006
Fresh fuel assembly misloaded outside of storage racks adjacent to Regions C and A	0.97107 ± 0.00050	0.00123
Heavy Load Accident in Region A (Reference k_{eff} = 0.97409 ± 0.00033)	1.07338 ± 0.00067	0.10029
Heavy Load Accident in Region B (Reference k_{eff} = 0.97628 ± 0.00033)	1.06936 ± 0.00070	0.09411
Heavy Load Accident in Region C (Reference k_{eff} = 0.97420 ± 0.00054)	1.00871 ± 0.00050	0.03555
Spent Fuel Temperatures from 150 to 248 degrees Fahrenheit(Reference k_{eff} at 150 degrees = 0.97955 ± 0.00052)	0.97773 ± 0.00051	-0.0008

Table 3.6-3
K_{eff} for Borated Accident Events

Description of Accident	Soluble Boron	K_{eff}
Fresh Fuel assembly misloaded in to a Region A storage cell that does not have a Boraflex poison box	200	0.96442 ± 0.00052
Fresh Fuel assembly misloaded in to a Region B storage cell that does not have a Boraflex poison box	400	0.96095 ± 0.00071
Fresh Fuel assembly misloaded in to a Region C storage cell	400	0.96047 ± 0.00038
Fresh fuel assembly misloaded between fresh fuel assembly in elevator and Region C	500	0.95145 ± 0.00068
Fresh fuel assembly misloaded outside of storage racks adjacent to Regions C and A	Not limiting	N/A
Heavy Load Accident in Region A	800	0.96425 ± 0.00063
Heavy Load Accident in Region A	1400	0.89844 ± 0.00061
Heavy Load Accident in Region B	800	0.96960 ± 0.00065
Heavy Load Accident in Region B	1400	0.91135 ± 0.00057
Heavy Load Accident in Region C	200	0.97736 ± 0.00051
Heavy Load Accident in Region C	1400	0.83773 ± 0.00047
Spent Fuel Temperatures from 150 to 248 degrees Fahrenheit	Not limiting	N/A

4 Summary of Results

The following sections contain the criticality analysis results for the Millstone Unit 2 spent fuel pool with soluble boron credit.

4.1 Allowable Storage Configurations

- Figure 4.1-1 displays the allowable storage configurations for Region A. Region A will be employed to store low burnup fuel assemblies which meet the requirements of Figure 4.2-1 in an "All-Cell" (4/4) storage configuration. Also, for future in-service testing, within any Region A module, two storage cells may contain no Boraflex poison box and no fuel assembly provided that the locations are separated by at least 3 storage cells.

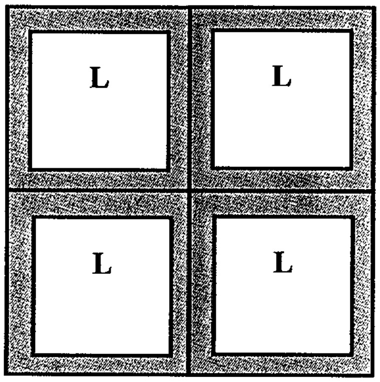
- Figure 4.1-2 displays the allowable storage configurations for Region C. Region C will be employed to store discharged fuel assemblies which meet the requirements of Figure 4.2-2 without the use of poison RODLETS. Region C will also be employed to store discharged fuel assemblies which meet the requirements of Figure 4.2-3 with three poison RODLETS stored in diagonal guide tube locations (one RODLET in the center guide and one RODLET in diagonally opposite locations). Fuel assemblies with poison RODLETs may be stored in Region C in any rotational orientation.

- Alternatively, Region C may be employed to store Consolidated Fuel Storage Boxes (CFSB). The fuel rods stored in a CFSB must meet the requirements of Figure 3.9-3 contained in Reference 1 (existing technical specification requirements).

- Region C may contain a mixture of CFSB's, and intact fuel assemblies with and without poison RODLETS, providing each CFSB and fuel assembly meets its Region C storage requirement described above.

- Figure 4.1-3 displays the allowable storage configurations for Region B. Note that Region B is employed to store fresh (or burned) fuel assemblies with a maximum initial enrichment (radially averaged) less than or equal to 4.85 w/o U-235 in a 3/4 storage configuration (three fresh/one Batch B fuel assembly). The Batch B fuel assembly may be stored in Region B cell locations which are presently blocked. The Batch B fuel assembly can have a maximum initial enrichment less than or equal to 2.36 w/o U-235 and an assembly burnup greater than or equal to 22,300 MWD/MT. All Batch B fuel assemblies currently residing in the Millstone 2 spent fuel pool meet these requirements.

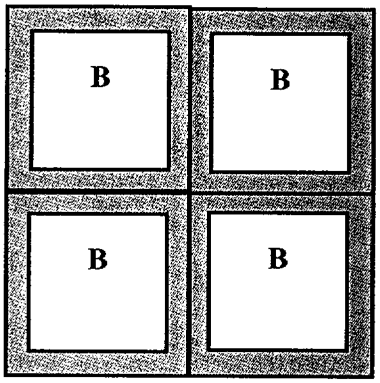
Figure 4.1-1
Allowable Storage Configuration in Region A



"L" represents

- Burned fuel assembly which meets the requirements of Figure 4.2-1.
- **<u>or</u>**
- An empty location.

Figure 4.1-2
Allowable Storage Configuration in Region C



"B" represents

- Burned fuel assembly which meets the requirements of Figure 4.2-2 for fuel without RODLETS or Figure 4.2-3 for fuel with RODLETS

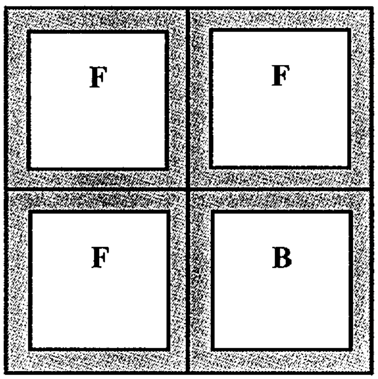
<u>or</u>

- Consolidated Fuel Storage Box (CFSB) which meets the requirements of existing Technical Specifications

  <u>or</u>

- An empty location

Figure 4.1-3
Allowable Storage Configuration in Region B



"F" represents

- Fresh or Burned fuel assembly which a initial enrichment less than or equal to 4.85 w/o ^{235}U

<u>**or**</u>

- An empty location

"B" represents

- Batch B Fuel Assembly with a maximum initial enrichment less than or equal to 2.36 w/o U-235 and a minimum assembly burnup equal to 22,300 MWD/MT

<u>**or**</u>

- An empty location

4.2 Burnup versus Enrichment Storage Requirements

- Figure 4.2-1 displays the burnup versus enrichment storage curve for Region A. The Region A requirements are tabulated in Table 4.2-1. Figure 4.2-1 contains the linear curve fit which describes the burnup required to store burned fuel assemblies in Region A as a function of initial enrichment.

- Figure 4.2-2 displays the burnup versus enrichment storage curve for Region C without the use of three poison RODLETS. This data is also tabulated in Table 4.2-2. Figure 4.2-2 also contains a third order polynomial which describes the burnup required to store burned fuel assemblies in Region C (without the use of three poison RODLETS) as a function of initial enrichment.

- Figure 4.2-3 displays the burnup versus enrichment storage curve for Region C with the use of three poison RODLETS located in diagonal guide tube locations (one RODLET in the center guide and one RODLET in diagonally opposite guide tube locations) This data is also tabulated in Table 4.2-2. Figure 4.2-3 also contains a third order polynomial which describes the burnup required to store burned fuel assemblies in Region C as a function of initial enrichment with three poison RODLETS installed as described above.

Table 4.2-1
Required Fuel Assembly Burnup (MWD/MT) versus Average Planar Initial Enrichment for the Region A Storage Cells

Enrichment	Burnup (MWD/MT)
3.64	0
5.00	7,683

Table 4.2-2
Required Fuel Assembly Burnup (MWD/MT) versus Average Planar Initial Enrichment for the Region C Storage Cells

Enrichment [wt. % U-235]	Burnup Required with no RODLETS [MWD/MT]	Burnup Required with RODLETS [MWD/MT]
1.1955	0	N/A
1.6250	N/A	0
3	31,490	19,445
4	46,093	32,129
5	60,667	44,479

Figure 4.2-1
Minimum Required Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region A

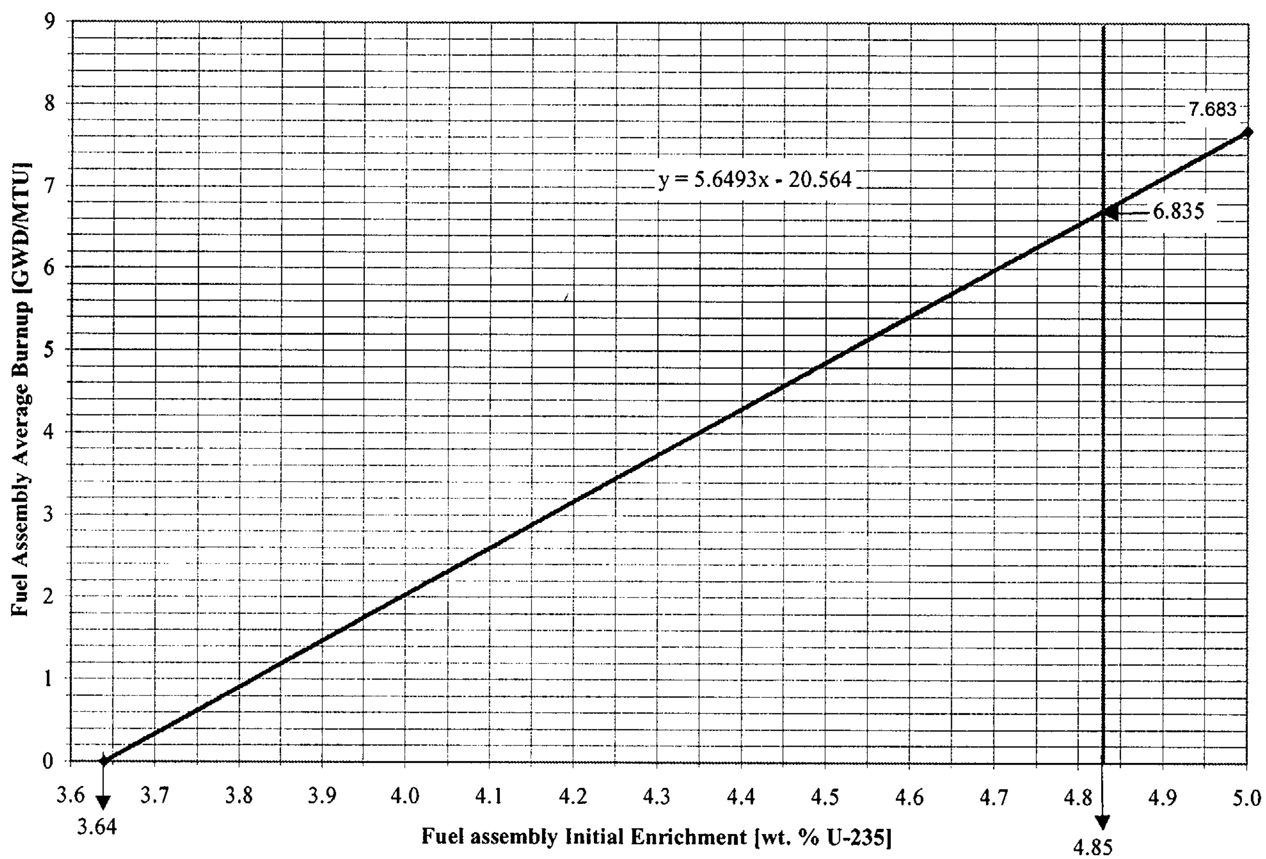


Figure 4.2-2
Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C for 4-out-of-4 fuel storage and no RODLETS.

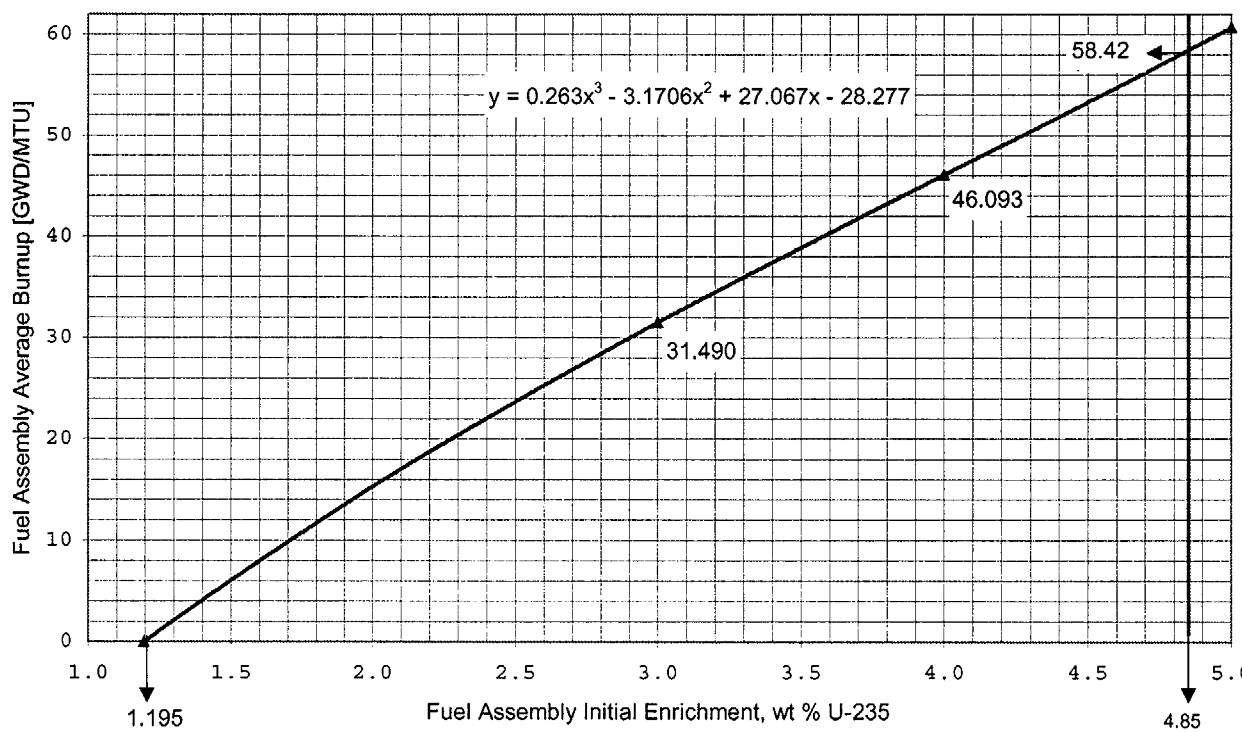
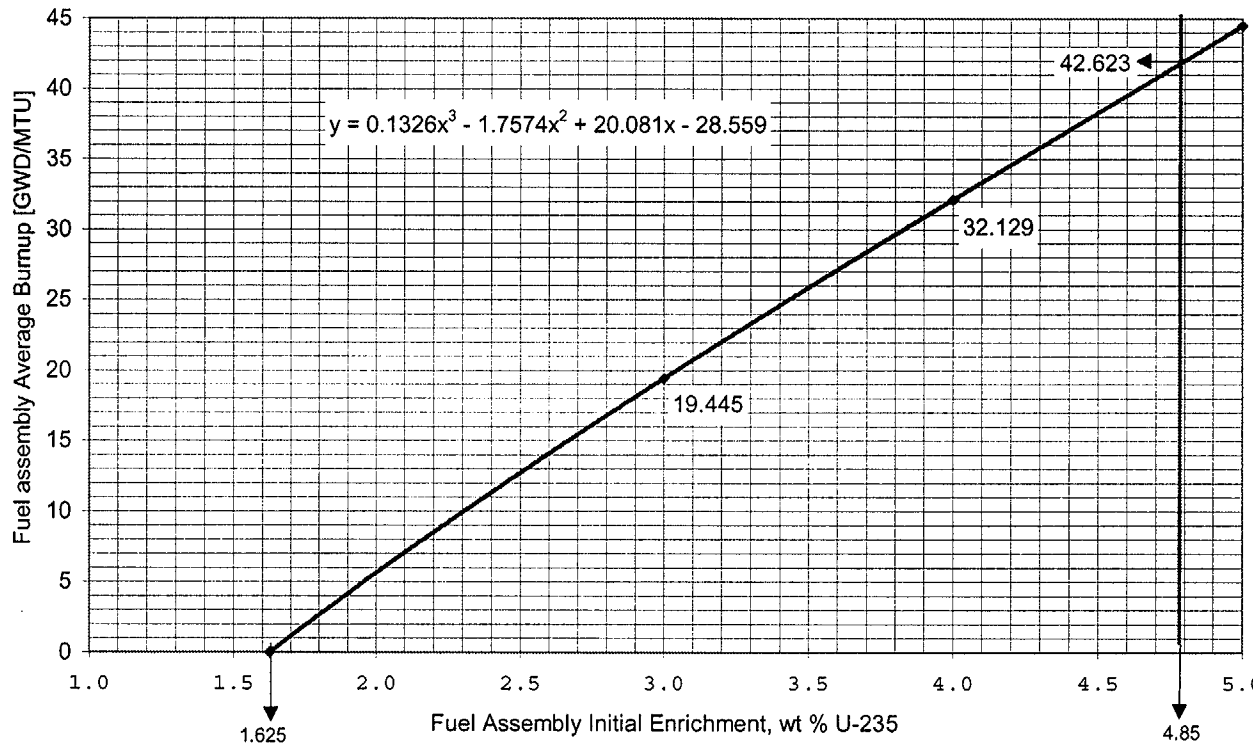


Figure 4.2-3
Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C for 4-out-of-4 fuel storage with RODLETS installed.



4.3 Total Soluble Boron Requirement

The total soluble boron (sum of all the three components) required to maintain the k_{eff} value (including all biases and uncertainties, without the adjustment for ^{10}B) less than or equal to 0.95 is determined to be 1358 ppm for a ^{10}B atom percent equal to 19.9. The soluble boron concentration required for a ^{10}B atom percent equal to 19.7 is 1372 ppm. The recommended minimum boron level is 1400 ppm and is sufficient to accommodate all the design requirements.

5 References

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18. S. Dembek (NRC) to H.A. Sepp (WEC), "Non-Conservatisms in Axial Burnup Biases for Spent Fuel Rack Criticality Analysis Methodology", July 27, 2001.

APPENDIX A.
Isotopic Number Densities employed in KENO calculations

Isotopic Number Densities used in KENO for the Four-Zone Model (3D)

```
'FUEL ZONE 1 3.0 WT.% 15,000 MWD/MT
SM-149    1  0    1.0711E-07                   END
U-235     1  0    3.8347E-04                   END
U-236     1  0    5.7439E-05                   END
U-238     1  0    2.2474E-02                   END
PU-239    1  0    1.0133E-04                   END
PU-240    1  0    2.4109E-05                   END
PU-241    1  0    1.2088E-05                   END
O-16      1  0    4.6874E-02                   END
B-10      1  0    0.97273E-5                   END
'FUEL ZONE 2 3.0 WT.% 15,000 MWD/MT
SM-149    2  0    1.0600E-07                   END
U-235     2  0    4.0194E-04                   END
U-236     2  0    5.4724E-05                   END
U-238     2  0    2.2488E-02                   END
PU-239    2  0    9.9596E-05                   END
PU-240    2  0    2.2472E-05                   END
PU-241    2  0    1.1083E-05                   END
O-16      2  0    4.6874E-02                   END
B-10      2  0    0.92710E-5                   END
'FUEL ZONE 3 3.0 WT.% 15,000 MWD/MT
SM-149    3  0    9.4135E-08                   END
U-235     3  0    4.6242E-04                   END
U-236     3  0    4.4436E-05                   END
U-238     3  0    2.2548E-02                   END
PU-239    3  0    8.6662E-05                   END
PU-240    3  0    1.5862E-05                   END
PU-241    3  0    6.8181E-06                   END
O-16      3  0    4.6874E-02                   END
B-10      3  0    0.73706E-5                   END
'FUEL ZONE 4 3.0 WT.% 15,000 MWD/MT
SM-149    4  0    8.0325E-08                   END
U-235     4  0    5.4384E-04                   END
U-236     4  0    3.0168E-05                   END
U-238     4  0    2.2617E-02                   END
PU-239    4  0    6.4591E-05                   END
PU-240    4  0    8.2061E-06                   END
PU-241    4  0    2.5991E-06                   END
O-16      4  0    4.6874E-02                   END
B-10      4  0    0.51255E-5                   END
```

```
'FUEL ZONE 1 3.0 WT.% 25,000 MWD/MT
SM-149     1   0     1.1136E-07                       END
U-235      1   0     2.4354E-04                       END
U-236      1   0     7.9322E-05                       END
U-238      1   0     2.2286E-02                       END
PU-239     1   0     1.1803E-04                       END
PU-240     1   0     4.2011E-05                       END
PU-241     1   0     2.3809E-05                       END
O-16       1   0     4.6874E-02                       END
B-10       1   0     1.49160E-5                       END
'FUEL ZONE 2 3.0 WT.% 25,000 MWD/MT
SM-149     2   0     1.1082E-07                       END
U-235      2   0     2.6637E-04                       END
U-236      2   0     7.6419E-05                       END
U-238      2   0     2.2313E-02                       END
PU-239     2   0     1.1818E-04                       END
PU-240     2   0     3.9722E-05                       END
PU-241     2   0     2.2496E-05                       END
O-16       2   0     4.6874E-02                       END
B-10       2   0     1.41737E-5                       END
'FUEL ZONE 3 3.0 WT.% 25,000 MWD/MT
SM-149     3   0     9.8702E-08                       END
U-235      3   0     3.4226E-04                       END
U-236      3   0     6.4585E-05                       END
U-238      3   0     2.2420E-02                       END
PU-239     3   0     1.0877E-04                       END
PU-240     3   0     2.9754E-05                       END
PU-241     3   0     1.5654E-05                       END
O-16       3   0     4.6874E-02                       END
B-10       3   0     1.12066E-5                       END
'FUEL ZONE 4 3.0 WT.% 25,000 MWD/MT
SM-149     4   0     8.3855E-08                       END
U-235      4   0     4.5352E-04                       END
U-236      4   0     4.5955E-05                       END
U-238      4   0     2.2541E-02                       END
PU-239     4   0     8.7867E-05                       END
PU-240     4   0     1.6772E-05                       END
PU-241     4   0     7.1961E-06                       END
O-16       4   0     4.6874E-02                       END
B-10       4   0     0.76222E-5                       END
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'FUEL ZONE 1 3.0 WT.% 35,000 MWD/MT
SM-149    1  0    1.1095E-07                     END
U-235     1  0    1.4670E-04                     END
U-236     1  0    9.1958E-05                     END
U-238     1  0    2.2081E-02                     END
PU-239    1  0    1.2207E-04                     END
PU-240    1  0    5.5837E-05                     END
PU-241    1  0    3.2126E-05                     END
O-16      1  0    4.6874E-02                     END
B-10      1  0    1.94246E-5                     END
'FUEL ZONE 2 3.0 WT.% 35,000 MWD/MT
SM-149    2  0    1.1094E-07                     END
U-235     2  0    1.6940E-04                     END
U-236     2  0    8.9759E-05                     END
U-238     2  0    2.2123E-02                     END
PU-239    2  0    1.2397E-04                     END
PU-240    2  0    5.3602E-05                     END
PU-241    2  0    3.1159E-05                     END
O-16      2  0    4.6874E-02                     END
B-10      2  0    1.86106E-5                     END
'FUEL ZONE 3 3.0 WT.% 35,000 MWD/MT
SM-149    3  0    9.9396E-08                     END
U-235     3  0    2.4801E-04                     END
U-236     3  0    7.9158E-05                     END
U-238     3  0    2.2284E-02                     END
PU-239    3  0    1.1895E-04                     END
PU-240    3  0    4.2324E-05                     END
PU-241    3  0    2.3787E-05                     END
O-16      3  0    4.6874E-02                     END
B-10      3  0    1.47685E-5                     END
'FUEL ZONE 4 3.0 WT.% 35,000 MWD/MT
SM-149    4  0    8.5103E-08                     END
U-235     4  0    3.7586E-04                     END
U-236     4  0    5.9068E-05                     END
U-238     4  0    2.2462E-02                     END
PU-239    4  0    1.0257E-04                     END
PU-240    4  0    2.5571E-05                     END
PU-241    4  0    1.2570E-05                     END
O-16      4  0    4.6874E-02                     END
B-10      4  0    1.00000E-5                     END
```

```
'FUEL ZONE 1 4.0 WT.% 25,000 MWD/MT
SM-149    1  0    1.3175E-07                       END
U-235     1  0    4.1474E-04                       END
U-236     1  0    9.4649E-05                       END
U-238     1  0    2.2105E-02                       END
PU-239    1  0    1.2497E-04                       END
PU-240    1  0    3.6013E-05                       END
PU-241    1  0    2.1201E-05                       END
O-16      1  0    4.6874E-02                       END
B-10      1  0    1.60717E-5                       END
'FUEL ZONE 2 4.0 WT.% 25,000 MWD/MT
SM-149    2  0    1.3196E-07                       END
U-235     2  0    4.4353E-04                       END
U-236     2  0    9.0642E-05                       END
U-238     2  0    2.2127E-02                       END
PU-239    2  0    1.2441E-04                       END
PU-240    2  0    3.3814E-05                       END
PU-241    2  0    1.9765E-05                       END
O-16      2  0    4.6874E-02                       END
B-10      2  0    1.54116E-5                       END
'FUEL ZONE 3 4.0 WT.% 25,000 MWD/MT
SM-149    3  0    1.2015E-07                       END
U-235     3  0    5.3877E-04                       END
U-236     3  0    7.4670E-05                       END
U-238     3  0    2.2222E-02                       END
PU-239    3  0    1.1164E-04                       END
PU-240    3  0    2.4526E-05                       END
PU-241    3  0    1.3005E-05                       END
O-16      3  0    4.6874E-02                       END
B-10      3  0    1.22701E-5                       END
'FUEL ZONE 4 4.0 WT.% 25,000 MWD/MT
SM-149    4  0    1.0556E-07                       END
U-235     4  0    6.7032E-04                       END
U-236     4  0    5.1463E-05                       END
U-238     4  0    2.2328E-02                       END
PU-239    4  0    8.6539E-05                       END
PU-240    4  0    1.3223E-05                       END
PU-241    4  0    5.4844E-06                       END
O-16      4  0    4.6874E-02                       END
B-10      4  0    0.84263E-5                       END
```

```
'FUEL ZONE 1 4.0 WT.% 35,000 MWD/MT
SM-149    1   0    1.2930E-07                    END
U-235     1   0    2.7993E-04                    END
U-236     1   0    1.1475E-04                    END
U-238     1   0    2.1925E-02                    END
PU-239    1   0    1.3263E-04                    END
PU-240    1   0    5.0347E-05                    END
PU-241    1   0    3.0762E-05                    END
O-16      1   0    4.6874E-02                    END
B-10      1   0    2.10373E-5                    END
'FUEL ZONE 2 4.0 WT.% 35,000 MWD/MT
SM-149    2   0    1.3035E-07                    END
U-235     2   0    3.1186E-04                    END
U-236     2   0    1.1098E-04                    END
U-238     2   0    2.1961E-02                    END
PU-239    2   0    1.3413E-04                    END
PU-240    2   0    4.7856E-05                    END
PU-241    2   0    2.9390E-05                    END
O-16      2   0    4.6874E-02                    END
B-10      2   0    2.01449E-5                    END
'FUEL ZONE 3 4.0 WT.% 35,000 MWD/MT
SM-149    3   0    1.1980E-07                    END
U-235     3   0    4.1963E-04                    END
U-236     3   0    9.4518E-05                    END
U-238     3   0    2.2102E-02                    END
PU-239    3   0    1.2596E-04                    END
PU-240    3   0    3.6300E-05                    END
PU-241    3   0    2.1144E-05                    END
O-16      3   0    4.6874E-02                    END
B-10      3   0    1.60896E-5                    END
'FUEL ZONE 4 4.0 WT.% 35,000 MWD/MT
SM-149    4   0    1.0625E-07                    END
U-235     4   0    5.7944E-04                    END
U-236     4   0    6.7572E-05                    END
U-238     4   0    2.2259E-02                    END
PU-239    4   0    1.0401E-04                    END
PU-240    4   0    2.0773E-05                    END
PU-241    4   0    1.0171E-05                    END
O-16      4   0    4.6874E-02                    END
B-10      4   0    1.10709E-5                    END
```

```
'FUEL ZONE 1 4.0 WT.% 45,000 MWD/MT
SM-149     1   0     1.2471E-07                      END
U-235      1   0     1.7953E-04                      END
U-236      1   0     1.2697E-04                      END
U-238      1   0     2.1731E-02                      END
PU-239     1   0     1.3292E-04                      END
PU-240     1   0     6.1458E-05                      END
PU-241     1   0     3.7234E-05                      END
O-16       1   0     4.6874E-02                      END
B-10       1   0     2.52892E-5                      END
'FUEL ZONE 2 4.0 WT.% 45,000 MWD/MT
SM-149     2   0     1.2635E-07                      END
U-235      2   0     2.1069E-04                      END
U-236      2   0     1.2423E-04                      END
U-238      2   0     2.1781E-02                      END
PU-239     2   0     1.3619E-04                      END
PU-240     2   0     5.9221E-05                      END
PU-241     2   0     3.6410E-05                      END
O-16       2   0     4.6874E-02                      END
B-10       2   0     2.45163E-5                      END
'FUEL ZONE 3 4.0 WT.% 45,000 MWD/MT
SM-149     3   0     1.1720E-07                      END
U-235      3   0     3.2027E-04                      END
U-236      3   0     1.0977E-04                      END
U-238      3   0     2.1976E-02                      END
PU-239     3   0     1.3276E-04                      END
PU-240     3   0     4.6952E-05                      END
PU-241     3   0     2.8272E-05                      END
O-16       3   0     4.6874E-02                      END
B-10       3   0     1.96923E-5                      END
'FUEL ZONE 4 4.0 WT.% 45,000 MWD/MT
SM-149     4   0     1.0541E-07                      END
U-235      4   0     4.9780E-04                      END
U-236      4   0     8.1587E-05                      END
U-238      4   0     2.2187E-02                      END
PU-239     4   0     1.1597E-04                      END
PU-240     4   0     2.8369E-05                      END
PU-241     4   0     1.5171E-05                      END
O-16       4   0     4.6874E-02                      END
B-10       4   0     1.33704E-5                      END
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'FUEL ZONE 1 4.0 WT% 55,000 MWD/MT
SM-149     1   0     1.2003E-07                            END
U-235      1   0     1.0893E-04                            END
U-236      1   0     1.3250E-04                            END
U-238      1   0     2.1521E-02                            END
PU-239     1   0     1.3017E-04                            END
PU-240     1   0     6.9171E-05                            END
PU-241     1   0     4.0910E-05                            END
O-16       1   0     4.6874E-02                            END
B-10       1   0     2.89831E-5                            END
'FUEL ZONE 2 4.0 WT% 55,000 MWD/MT
SM-149     2   0     1.2188E-07                            END
U-235      2   0     1.3622E-04                            END
U-236      2   0     1.3138E-04                            END
U-238      2   0     2.1589E-02                            END
PU-239     2   0     1.3460E-04                            END
PU-240     2   0     6.7632E-05                            END
PU-241     2   0     4.0847E-05                            END
O-16       2   0     4.6874E-02                            END
B-10       2   0     2.80797E-5                            END
'FUEL ZONE 3 4.0 WT% 55,000 MWD/MT
SM-149     3   0     1.1357E-07                            END
U-235      3   0     2.3875E-04                            END
U-236      3   0     1.2088E-04                            END
U-238      3   0     2.1841E-02                            END
PU-239     3   0     1.3495E-04                            END
PU-240     3   0     5.6052E-05                            END
PU-241     3   0     3.3913E-05                            END
O-16       3   0     4.6874E-02                            END
B-10       3   0     2.29165E-5                            END
'FUEL ZONE 4 4.0 WT% 55,000 MWD/MT
SM-149     4   0     1.0357E-07                            END
U-235      4   0     4.2452E-04                            END
U-236      4   0     9.3668E-05                            END
U-238      4   0     2.2112E-02                            END
PU-239     4   0     1.2388E-04                            END
PU-240     4   0     3.5718E-05                            END
PU-241     4   0     2.0036E-05                            END
O-16       4   0     4.6874E-02                            END
B-10       4   0     1.57619E-5                            END
```

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'FUEL ZONE 1 5.0 WT.% 45,000 MWD/MT
SM-149    1   0     1.4446E-07                        END
U-235     1   0     3.1010E-04                        END
U-236     1   0     1.5174E-04                        END
U-238     1   0     2.1573E-02                        END
PU-239    1   0     1.4459E-04                        END
PU-240    1   0     5.7147E-05                        END
PU-241    1   0     3.6691E-05                        END
O-16      1   0     4.6874E-02                        END
B-10      1   0     2.74742E-5                        END
'FUEL ZONE 2 5.0 WT.% 45,000 MWD/MT
SM-149    2   0     1.4731E-07                        END
U-235     2   0     3.5112E-04                        END
U-236     2   0     1.4722E-04                        END
U-238     2   0     2.1617E-02                        END
PU-239    2   0     1.4749E-04                        END
PU-240    2   0     5.4575E-05                        END
PU-241    2   0     3.5369E-05                        END
O-16      2   0     4.6874E-02                        END
B-10      2   0     2.64200E-5                        END
'FUEL ZONE 3 5.0 WT.% 45,000 MWD/MT
SM-149    3   0     1.3934E-07                        END
U-235     3   0     4.9175E-04                        END
U-236     3   0     1.2620E-04                        END
U-238     3   0     2.1791E-02                        END
PU-239    3   0     1.4092E-04                        END
PU-240    3   0     4.1805E-05                        END
PU-241    3   0     2.6080E-05                        END
O-16      3   0     4.6874E-02                        END
B-10      3   0     2.13113E-5                        END
'FUEL ZONE 4 5.0 WT.% 45,000 MWD/MT
SM-149    4   0     1.2837E-07                        END
U-235     4   0     7.0210E-04                        END
U-236     4   0     9.0582E-05                        END
U-238     4   0     2.1980E-02                        END
PU-239    4   0     1.1844E-04                        END
PU-240    4   0     2.4170E-05                        END
PU-241    4   0     1.2948E-05                        END
O-16      4   0     4.6874E-02                        END
B-10      4   0     1.47966E-5                        END
```

```
'FUEL ZONE 1 5.0 WT.% 55,000 MWD/MT
SM-149    1  0    1.2003E-07                     END
U-235     1  0    1.0893E-04                     END
U-236     1  0    1.3250E-04                     END
U-238     1  0    2.1521E-02                     END
PU-239    1  0    1.3017E-04                     END
PU-240    1  0    6.9171E-05                     END
PU-241    1  0    4.0910E-05                     END
O-16      1  0    4.6874E-02                     END
B-10      1  0    2.89831E-5                     END
'FUEL ZONE 2 5.0 WT.% 55,000 MWD/MT
SM-149    2  0    1.2188E-07                     END
U-235     2  0    1.3622E-04                     END
U-236     2  0    1.3138E-04                     END
U-238     2  0    2.1589E-02                     END
PU-239    2  0    1.3460E-04                     END
PU-240    2  0    6.7632E-05                     END
PU-241    2  0    4.0847E-05                     END
O-16      2  0    4.6874E-02                     END
B-10      2  0    2.80797E-5                     END
'FUEL ZONE 3 5.0 WT.% 55,000 MWD/MT
SM-149    3  0    1.1357E-07                     END
U-235     3  0    2.3875E-04                     END
U-236     3  0    1.2088E-04                     END
U-238     3  0    2.1841E-02                     END
PU-239    3  0    1.3495E-04                     END
PU-240    3  0    5.6052E-05                     END
PU-241    3  0    3.3913E-05                     END
O-16      3  0    4.6874E-02                     END
B-10      3  0    2.29165E-5                     END
'FUEL ZONE 4 5.0 WT.% 55,000 MWD/MT
SM-149    4  0    1.0357E-07                     END
U-235     4  0    4.2452E-04                     END
U-236     4  0    9.3668E-05                     END
U-238     4  0    2.2112E-02                     END
PU-239    4  0    1.2388E-04                     END
PU-240    4  0    3.5718E-05                     END
PU-241    4  0    2.0036E-05                     END
O-16      4  0    4.6874E-02                     END
B-10      4  0    1.57619E-5                     END
```

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'FUEL ZONE 1 5.0 WT.% 65,000 MWD/MT
SM-149    1  0    1.3018E-07                 END
U-235     1  0    1.3160E-04                 END
U-236     1  0    1.6882E-04                 END
U-238     1  0    2.1184E-02                 END
PU-239    1  0    1.3804E-04                 END
PU-240    1  0    7.2808E-05                 END
PU-241    1  0    4.4541E-05                 END
O-16      1  0    4.6874E-02                 END
B-10      1  0    3.45894E-5                 END
'FUEL ZONE 2 5.0 WT.% 65,000 MWD/MT
SM-149    2  0    1.3376E-07                 END
U-235     2  0    1.6676E-04                 END
U-236     2  0    1.6760E-04                 END
U-238     2  0    2.1260E-02                 END
PU-239    2  0    1.4387E-04                 END
PU-240    2  0    7.1412E-05                 END
PU-241    2  0    4.4759E-05                 END
O-16      2  0    4.6874E-02                 END
B-10      2  0    3.37491E-5                 END
'FUEL ZONE 3 5.0 WT.% 65,000 MWD/MT
SM-149    3  0    1.2876E-07                 END
U-235     3  0    3.0052E-04                 END
U-236     3  0    1.5394E-04                 END
U-238     3  0    2.1544E-02                 END
PU-239    3  0    1.4647E-04                 END
PU-240    3  0    5.9242E-05                 END
PU-241    3  0    3.7517E-05                 END
O-16      3  0    4.6874E-02                 END
B-10      3  0    2.80084E-5                 END
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SM-149    4  0    1.2276E-07                 END
U-235     4  0    5.3951E-04                 END
U-236     4  0    1.1838E-04                 END
U-238     4  0    2.1843E-02                 END
PU-239    4  0    1.3567E-04                 END
PU-240    4  0    3.7520E-05                 END
PU-241    4  0    2.2172E-05                 END
O-16      4  0    4.6874E-02                 END
B-10      4  0    1.94600E-5                 END
```

APPENDIX B.
Boron-10 Number Densities Employed in KENO Calculations

Enrichment = 3.000 w/o U-235, Burnup = 15000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	1.19453	0.97273	1.19423 ± 0.00045
2	1.20684	0.92710	1.20684 ± 0.00046
3	1.24483	0.73706	1.24503 ± 0.00047
4	1.29297	0.51255	1.29307 ± 0.00047

Enrichment = 3.000 w/o U-235, Burnup = 25000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	1.09174	1.49160	1.09166 ± 0.00044
2	1.11140	1.41737	1.11141 ± 0.00043
3	1.16796	1.12066	1.16771 ± 0.00046
4	1.24043	0.76222	1.24075 ± 0.00046

Enrichment = 3.000 w/o U-235, Burnup = 35000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	1.00010	1.94246	0.99978 ± 0.00041
2	1.02571	1.86106	1.02544 ± 0.00041
3	1.09704	1.47685	1.09707 ± 0.00044
4	1.19141	1.00000	1.19122 ± 0.00047

Enrichment = 4.000 w/o U-235, Burnup = 25000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	1.18273	1.60717	1.18250 ± 0.00046
2	1.19990	1.54116	1.19967 ± 0.00046
3	1.25222	1.22701	1.25265 ± 0.00047
4	1.31804	0.84263	1.31787 ± 0.00049

Enrichment = 4.000 w/o U-235, Burnup = 35000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	1.09298	2.10373	1.09258 ± 0.00045
2	1.11716	2.01449	1.11679 ± 0.00043
3	1.18678	1.60896	1.18704 ± 0.00046
4	1.27393	1.10709	1.27349 ± 0.00047

Enrichment = 4.000 w/o U-235, Burnup = 45000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	1.00854	2.52892	1.00813 ± 0.00041
2	1.03864	2.45163	1.03843 ± 0.00040
3	1.12349	1.96923	1.12318 ± 0.00045
4	1.23165	1.33704	1.23159 ± 0.00048

Enrichment = 4.000 w/o U-235, Burnup = 55000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	0.93464	2.89831	0.93502 ± 0.00040
2	0.96780	2.80797	0.96795 ± 0.00040
3	1.06241	2.29165	1.06249 ± 0.00044
4	1.19031	1.57619	1.18994 ± 0.00045

Enrichment = 5.000 w/o U-235, Burnup = 45000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	1.08766	2.74742	1.08727 ± 0.00042
2	1.11552	2.64200	1.11539 ± 0.00044
3	1.19582	2.13113	1.19626 ± 0.00046
4	1.29426	1.47966	1.29427 ± 0.00047

Enrichment = 5.000 w/o U-235, Burnup = 55000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	1.00980	3.13695	1.00949 ± 0.00041
2	1.04344	3.02802	1.04316 ± 0.00041
3	1.13883	2.48209	1.13871 ± 0.00043
4	1.25720	1.69486	1.25762 ± 0.00044

Enrichment = 5.000 w/o U-235, Burnup = 65000 MWD/MT

Zone No.	K_{inf} (DIT)	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (KENO)
1	0.93916	3.45894	0.93914 ± 0.00038
2	0.97626	3.37491	0.97611 ± 0.00038
3	1.08260	2.80084	1.08266 ± 0.00041
4	1.22053	1.94600	1.22056 ± 0.00045

Docket No. 50-336
B18501

Attachment 6

Millstone Power Station, Unit No. 2

Technical Specifications Change Request 2-10-01
Fuel Pool Requirements
Boron Dilution Analysis

Millstone Unit 2
Spent Fuel Pool Boron Dilution Analysis
Summary

Table of Contents

Total Pages: 18

1.0 INTRODUCTION

A Spent Fuel Pool (SFP) Criticality reanalysis has been completed for crediting soluble boron in the Millstone Unit 2 (MP2) SFP under both normal and accident conditions.

The purpose of that criticality reanalysis is to reflect the following:

- increase allowable fuel enrichment from 4.5 weight percent (w/o) to 4.85 w/o for all regions
- reduce Boraflex reactivity credit in Region A & B
- allow fuel to be located in the 40 cell blocked Region B locations
- credit soluble boron for normal conditions as well as accident conditions

As a result of the soluble boron credit for normal conditions, a boron dilution analysis is required and is presented in this Technical Evaluation (TE). This TE includes the following plant specific features and potential events:

- instrumentation
- administrative procedures
- boration sources
- dilution sources
- dilution flow rates
- boron dilution initiating events
- boron dilution times and volumes

This boron dilution analysis ensures that sufficient time is available to detect and mitigate the dilution before the design basis limit on the effective multiplication factor ($k_{eff} = 0.95$) is reached.

2.0 SPENT FUEL POOL AND RELATED SYSTEM FEATURES

This section provides background information on the SFP and related systems.

2.1 Spent Fuel Pool

The SFP is located in the auxiliary building. The pool is designed for the underwater storage of spent fuel assemblies after removal from the reactor core. The spent fuel storage racks, located in the bottom of the SFP are licensed to accommodate fuel assemblies in both intact and consolidated forms. The SFP is designed to maintain approximately 24 feet of borated water above the stored fuel assemblies.

2.2 Spent Fuel Pool Storage Racks

There are 3 Regions for fuel storage, designated Regions A, B and C. The stainless steel storage racks consisting of vertical cells grouped in parallel rows, are designed for a center-to-center distance of 9.8 inches in Regions A and B and 9.0 inches in Region C. Spent fuel decay heat is removed by the SFP cooling system described below. The fuel storage racks are designed to the requirements defined below while maintaining a physical arrangement that results in a K_{eff} of 0.95 or less during all normal usage of the racks and under abnormal conditions. The arrangement also provides for adequate convective cooling of stored fuel assemblies.

Region A consists of one 8 x 10 module and two 8 x 9 modules of poisoned spent fuel racks with a nominal center-to-center cell spacing of 9.8 inches. These modules are used to store spent fuel bundles which have achieved a specified burnup, or low enrichment fresh fuel. These modules can store up to 224 spent fuel bundles.

Region B consists of two 8 x 10 modules (160 total storage cells) with cell blockers installed in 40 locations to permit the unrestricted storage of fuel assemblies in 120 locations. The region consists of poisoned spent fuel racks with a nominal center-to-center cell spacing of 9.8 inches and can currently store up to 120 spent fuel bundles in the locations. It is proposed to store an additional 40 low reactivity fuel assemblies in the cell blocker locations.

Region C consists of fourteen modules of non-poisoned spent fuel racks with nominal center-to-center cell spacing of 9.0 inches. These modules are used to store spent fuel bundles that have achieved a specified fuel burnup. Depending on the amount of fuel burnup, neutron poison RODLETs may also be needed. The region consists of 962 cells, licensed for storage of both consolidated and intact fuel.

The fuel storage racks displace approximately 600 ft^3, and the fuel assembly volume displaced is 5,384 ft^3 assuming fuel in all storage locations. The volume of water in the SFP with the SFP at the low level setpoint, is 29,318 ft^3 or 219,314 gallons of water. No credit is taken here for the volume of water in the fuel transfer canal and cask laydown pit, which are connected to the SFP. These volumes of water are normally connected to the SFP, but they are conservatively ignored here.

2.3 Spent Fuel Pool Cooling

The function of the SFP cooling system is to remove decay heat generated by spent fuel assemblies stored in the pool by limiting the temperature of the borated pool water to an acceptable level, thereby ensuring the cladding integrity of stored spent fuel assemblies. The SFP cooling system consists of 2 trains of SFP cooling, which can be augmented by the shutdown cooling system during refueling outages. The SFP cooling system and shutdown cooling system are cooled by the Reactor Building Component Cooling Water (RBCCW) System.

2.4 Spent Fuel Pool Instrumentation

The SFP is provided with level and temperature instruments which provide annunciation in the main control room. The level alarm of the SFP will initiate the operators response to a potential boron dilution event in the SFP. The setpoint of this alarm will ensure the TS minimum is maintained. The SFP low level alarm is set for a water depth of 38 feet. The high level alarm is set for a water depth of 39'-2". The high level alarm is about 1 foot below the SFP operating deck. If the pool level were to be raised from the low level alarm point to the high level alarm point, a dilution of approximately 8,108 gallons could occur before the alarm is received in the control room.

SFP temperature instrumentation provides continuous monitoring (high temperature alarm) and recording of pool water temperatures by main control room personnel. A low-flow alarm will alert operating personnel that one or both SFP cooling water pumps has failed to operate. SFP cooling water flow instrumentation annunciates a low SFP cooling water flow alarm in the main control room. SFP heat exchanger outlet instrumentation annunciates a high-temperature alarm in the main control room.

2.5 Spent Fuel Pool Administrative Procedures

Currently, Technical Specifications (TS) requires the soluble boron concentration in the SFP to be greater than or equal to 800 ppm prior to moving a fuel assembly, consolidated fuel storage box or shielded cask. Chemistry practice maintains the SFP greater than or equal to 1720 ppm since that is the Refueling Water Storage Tank minimum requirement according to TS. Actual SFP boron concentration typically is maintained at about 2100 ppm. TS requirements are proposed to be changed to require the soluble boron concentration to be maintained ≥ 1720 ppm.

Operations Standard procedure, directs the requirement for plant equipment operator (PEO) rounds and specifies that, *'PEO rounds should be completed during the first part of the shift in order to become familiar with the condition and status of plant equipment.'*. PEO Rounds procedure states, *'At least twice each shift, unless specified otherwise, items and areas specified on applicable OPS Form are inspected by qualified PEOs.'*.

The location of the SFP operating deck is in the Unit 2 Auxiliary Building 38'-6" elevation. PEO rounds in the Unit 2 Auxiliary Building 38'-6" elevation requires a General Visual Inspection be performed twice each 12 hour shift and the SFP level be checked within the second 6 hours of each 12 hour shift. Also, with the requirement to begin and complete rounds in the first part of a shift, it is conceivable that the PEO rounds be completed in the first hour of the first half of a shift and the last hour of the second half of a shift. Therefore, the greatest time between PEO rounds on the SFP floor could be 12 hours.

2.6 Boration Sources

The normal source of borated water to the SFP is from the refueling water storage tank (RWST). The boron concentration in the RWST is maintained above 1720 ppm in accordance with TS Surveillance Requirement.

3.0 SPENT FUEL POOL DILUTION EVENT

3.1 Calculation of Boron Dilution Times and Volumes

The boron concentration in the SFP is proposed to be maintained greater than or equal to the proposed TS lower limit of 1720 ppm. Based on the criticality analysis, a soluble boron concentration of 600 ppm will maintain the reactivity within the design basis limit of $k_{eff} \leq 0.95$ (including uncertainties and burnup) with a 95% probability at a 95% confidence level for non-accident conditions. The possibility of a boron dilution accident must be considered. It must be shown that it is not credible to dilute from the proposed TS required boron concentration of 1720 ppm to less than 600 ppm of soluble boron. Less than 600 ppm of soluble boron could cause SFP k_{eff} to exceed 0.95 (including uncertainties and biases). It should be noted that even if SFP soluble boron concentration was to go to 0 ppm, SFP k_{eff} would be less than 1.00 (including uncertainties and biases). It should be noted that for accident conditions, up to 1400 ppm soluble boron is credited in the criticality analysis. However, consideration of a simultaneous occurrence of 2 unlikely and independent events such as a boron dilution event and another independent accident condition is not required to be considered by the double contingency principal.

The dilution times and volumes calculated are conservatively based on maintaining a final concentration of 600 ppm of soluble boron in the SFP. The total amount of unborated water that can be added to the SFP to reduce the boron concentration to 600 ppm needs to be determined.

There is no automatic SFP level control system in the SFP, so that any dilution to the SFP will add water to the SFP. Therefore, the addition of unborated water to the SFP will lead to increased SFP water level, and if not controlled, an overflow of the SFP. The method used to analyze this situation will be the continuous dilution method (feed and bleed). The continuous dilution method assumes unborated water is added at a constant rate with a constant rate of removal. This physically corresponds to unborated water being added to the SFP, and borated water at the current concentration being lost by overflow of the SFP. This feed and bleed method will give conservative results even if the dilution is initially a batch dilution as SFP level is raised. For conservative results, it is also assumed that, the initial SFP water volume will exclude the cask laydown canal, transfer canal, transfer tube and gate areas, and also exclude the water volume displaced by the fuel and fuel racks.

Excluding the cask laydown area, the gate areas, transfer canal and transfer tube and by conservatively only crediting the SFP water level is at the SFP level low alarm setpoint, the calculated SFP volume is 264,077 gallons. The volume displaced from all the fuel racks is approximately 600 ft^3 and the fuel assembly volume displaced is 5,384 ft^3. The volume displaced by the fuel assumes fuel is stored in all rack locations. Therefore, subtracting the fuel and the rack volume will yield a SFP total water volume remaining of 219,314 gallons.

A continuous "feed and bleed" dilution of the SFP will yield the most conservative results in this boron dilution analysis. This will be calculated by the equation for change in boron mass per unit of time.

$$dm/dt = m_{in}^{o} - m_{out}^{o}$$

where m_{in}^{o} , m_{out}^{o} are the mass flow rates of boron in and out, respectively.

Ignoring the minimal temperature effects, the mass flow rate of boron in each instance is equal to the product of the volumetric flow rate of diluted water, Q, and the concentration of boron, C, within the diluted water.

$$m^{o} = Q * C$$

If the concentration of water volume added is zero and the flow rate out is equal to the flow rate in, the equation can be rewritten as:

$$dm/dt = -Q_{out} * C_{out} = V_{SFP\ Total} * dC/dt$$

where:

$V_{SFP\ Total}$ = volume of the SFP, at the low level alarm
dC/dt = change in concentration of the SFP with respect to time

Therefore, if the equation above is rearranged and integrated from zero to time (t), the following would be the result:

$$-Q_{out} * t = V_{SFP\ Total} * (\ln C_t - \ln C_o)$$

Realizing that Q is equal to the volume divided by time and that the volume out is equal to the volume in, the left side of the equation reduces to the negative of the volume in V_{in} . Then, by moving the negative on the left to the right and realizing that C_t is equal to our final concentration and C_o is our initial concentration, the above equation can be rewritten in its final form.

$$V_{in} = V_{SFP\ Total} * \ln (C_t / C_o)$$

The SFP volume at the low level alarm, less the volume displaced by the racks and fuel, is 219,314 gallons. Using this volume, we can determine the total dilution volume needed to dilute the SFP from the initial 1720 ppm boron concentration (TS limit) to the minimum acceptable 600 ppm, by solving the above equation for $\mathbf{V_{in}}$.

Inserting the following values:
$V_{SFP\ Total}$ = 219,314 gallons
C_t = 1720 ppm
C_o = 600 ppm

TS Limit

$$\mathbf{V_{in} = 219{,}314\ gals * \ln(1720\ ppm / 600\ ppm)}$$
$$\mathbf{V_{in} = 230{,}971\ gallons}$$

The result is 230,971 gallons. Therefore, any dilution source not capable of supplying 230,971 gallons of unborated water will not be capable of diluting the pool to 600 ppm from a starting value of 1720 ppm.

For dilution sources with automatic make-up, the capacity for dilution is essentially infinite. Should one of these sources begin adding unborated water to the pool, the pool level would rise to the high level alarm setpoint, alerting the control room operators. Should the high level alarm fail, and no plant equipment operator (PEO) actions were taken, the pool will eventually fill to the curb and begin overflowing. The effects of this overflow would be apparent to the PEO's performing their rounds. 12 hours is the conservatively longest interval between PEO rounds to detect the SFP overflow. Assuming a arbitrary dilution flow rate of 200 gpm of unborated water, more than 19 hours are needed for the SFP soluble boron concentration to change from 1720 ppm to 600 ppm. This is simply calculated as follows:

230,971 gallons / 200 gpm = 1155 minutes = conservatively about 19 hours

Since the conservatively longest time between PEO rounds is 12 hours, and 19 hours are needed at 200 gpm, to dilute the SFP soluble boron concentration to 600 ppm, there is ample time to detect and secure the dilution event.

In summary, the ability to prevent the SFP soluble boron concentration from being diluted from the TS minimum value of 1720 ppm to a value of 600 ppm will be shown to meet one of the following two criteria :

- Any dilution source not capable of supplying 230,971 gallons of unborated water will not be capable of diluting the pool to 600 ppm from a starting value of 1720 ppm.

- If the dilution flowrate of unborated water is $\leq$ 200 gpm, then at least 19 hours will be needed for the SFP soluble boron concentration to be reduced from 1720 ppm to 600 ppm. All dilution scenarios evaluated here will eventually cause a SFP high water level alarm in the control room, and as a back-up, the Plant Equipment Operator (PEO) would detect high SFP water levels or SFP overflow. Since the conservatively longest time between PEO rounds is 12 hours, and 19 hours, as shown above, are needed at 200 gpm to dilute the SFP soluble boron concentration to 600 ppm, there is ample time to detect and terminate the dilution event.

4.0 DILUTION SOURCE PATH EVALUATION

This section evaluates the potential for dilution of the SFP both from the SFP Cooling System as well as from external sources within the SFP building.

4.1 Spent Fuel Pool Cooling

There is limited potential for addition of water from systems that cross-connect into the SFP Cooling system. Potential water addition can be supplied from the Low Pressure Safety Injection (LPSI) system through the Shutdown Cooling Heat Exchangers. This system is capable of injecting 3000 gpm for approximately 15 minutes from a borated water source. That borated water source being the Refueling Water Storage Tank (RWST) which is maintain at or above 1720 ppm boron. The RWST is also the source of water for the Refueling Pool Purification system at a transfer rate of 125 gpm. Both sources are isolated by a multiple of normally closed valves, 2-SI-458, 2-RW-15 and 2-RW-27 and either 2-RW-25 or 2-RW-28B depending which RW Purification Pump is in operation, and controlled procedurally by Operations. This LPSI system is not considered a threat to dilute the SFP boron concentration to 600 ppm since the injected water is from a borated water source with a concentration ≥ 1720 ppm.

4.2 Auxiliary Feedwater

The Auxiliary Feedwater (AFW) system takes suction from the Condensate Storage Tank (CST) and is a backup supply of makeup water with a flow rate of 100 gpm to the SFP. It is isolated by a normally locked closed valve, 2-FW-54, and controlled procedurally by Operations. The CST is a non-borated water source with a useable volume of 250,000 gallons. Makeup to the CST is a manual evolution that is performed by Operations and controlled by procedure. Auxiliary Feedwater is not considered a dilution which will threaten reaching a SFP soluble boron concentration of 600 ppm, since the flow rate of 100 gpm is less than the 200 gpm dilution flow rate of unborated water needed to dilute the SFP from 1720 ppm to 600 ppm in 19 hours. Operators would be alerted to this event by a high SFP water level alarm, or PEO rounds identifying a high SFP water level or SFP overflow condition. Therefore, a leak in the Auxiliary Feedwater system is not considered a threat to dilute the SFP below 600 ppm.

4.3 Primary Makeup Water

The Primary Makeup Water system is the normal makeup water supply to the SFP from the Primary Water Storage Tank (PWST). This is being supplied at a minimum rate of 50 gpm which is adequate for the water loss due to evaporation and any system leakage that may occur. The maximum makeup capability of this permanently installed system is 200 gpm. This manipulation is a manual evolution that is performed by Operations and controlled by procedure. This is not considered a dilution which will threaten reaching a SFP boron concentration of 600 ppm since the PWST has a capacity of 150,000 gallons, and if its contents were to be discharged into the SFP, it is less than the 230,971 gallons needed to reach a SFP soluble boron concentration of 600 ppm. Makeup to the PWST is a manual evolution and performed by Operations and controlled by procedure. Operators would be alerted to this event by a high SFP water level alarm, or PEO rounds identifying a high SFP water level or SFP overflow condition.

4.4. RBCCW

The Reactor Building Closed Cooling Water (RBCCW) system provides coolant to the shell side of the SFP heat exchangers. The heat exchanger tubes form a physical barrier between the RBCCW and the SFP cooling systems. The pressures on the tube side and the shell side are nearly equal, with the RBCCW pressure slightly higher. If a tube leak were to occur, the RBCCW would enter the SFP cooling system, diluting the pool. The volume of the RBCCW is approximately 42,000 gallons, however, makeup to the RBCCW system surge tank is from the Primary Water System. The Primary Makeup Water (PMW) System has a 150,000 gallon tank capacity and the PMW makeup pumps are capable of providing 200 gpm. Since the combined PMW and RBCCW system volumes (42,000 + 150,000 = 192,000 gallons) is less than the 230,971 gallons, there is not sufficient unborated water to dilute the SFP to a soluble boron concentration of 600 ppm. Operators would be alerted to this event by a high SFP water level alarm, or Plant Equipment Operator rounds identifying a high SFP water level or SFP overflow condition.

4.5. Filling the Transfer Canal

The transfer canal is normally full of borated water and open to the SFP. If it was empty and needed to be filled, the transfer canal fill is accomplished by way of a batch process using water from the SFP. This process is controlled procedurally by Operations and is currently performed by placing a submersible pump in the SFP, then raising the SFP level using water from the RWST (a borated water supply) to below the high level alarm in the SFP. Then the fill to the SFP is secured and the submersible pump is started, pumping SFP water to the transfer canal. This process is repeated until the transfer canal is at the desirable level. Once the fill process is completed, the canal bulkhead gate can be opened to equalize the two areas in support of refueling operations. Using this process, only borated water is used.

Even in the unlikely event that a unborated water source was used to fill the transfer canal, the volume is not sufficient to dilute the SFP to 600 ppm. The transfer canal has a capacity of 76,387 gallons, which when combined with the small volumes within the transfer tube and bulkhead gate areas equals 78,405 gallons. This volume is less than the 230,971 gallons needed to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm.

4.6. Filling the Cask Laydown Pit

The cask laydown pit is adjacent to the SFP and is isolated by a bulkhead gate. A spent fuel shipping cask can be placed in the SFP cask laydown area for loading of fuel. The cask laydown pit is filled from the SFP itself or the RWST, which is a borated water source. The volume of the cask laydown pit is 23,326 gallons. This volume is less than the 230,971 gallons needed to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm .

5.0 PIPE BREAKS AND LEAKS

5.1 Pipe Break/Leak Methodology

In order to address the potential for a dilution event in the MP2 SFP, the general area on the Auxiliary Building 38'- 6" elevation (location of SFP operating deck) was inspected. This inspection revealed several piping systems in the general vicinity. The mechanical attributes and operating parameters of these piping systems are utilized to determine the proper application of pipe break rules.

Of the piping systems identified, only the Auxiliary Steam & Condensate Return were determined to meet the high energy line classification (piping systems with normal operating temperature equal to or greater than 200 °F, or normal operating pressure equal to or greater than 275 psig). The balance of piping systems are below these thresholds and are therefore defined as moderate energy.

Moderate Energy Pipe Break/Crack Postulation

The MP2 FSAR was reviewed to determine the applicable rules for evaluating moderate energy piping for pipe breaks or cracks. FSAR Section 6.1.4.1 provides the requirements for moderate energy piping and specifies that no cracks are required to be postulated regardless of seismic design. The HELB Program Manual is consistent with this position. Specifically, no pipe breaks or cracks are required to be postulated in moderate energy systems. Furthermore, postulated piping failures are non-mechanistic, and therefore not caused by a seismic event. Therefore, the fact that the piping systems evaluated are not designed to seismic standards is not pertinent to this evaluation. Based on the current licensing requirements for MP2, no piping breaks or cracks are required to be postulated for moderate energy piping systems in the vicinity of the SFP.

The Hazards Program makes reference to the Standard Review Plan (SRP) NUREG-0800 for guidance in assessing postulated pipe breaks and cracks. Although the licensing basis for Unit 2 does not invoke the SRP, if the SRP rules were applied, the following assessment is provided.

The rules governing postulation of pipe breaks in the SRP are provided in Sections 3.6.1 and 3.6.2, promulgated by the NRC Auxiliary Systems Branch (ASB) and Mechanical Engineering Branch (MEB), respectively. The SRP Sections 3.6.1 is generally concerned with the effects of pipe breaks on essential systems and the impact on safe shutdown capability. Sections 3.6.2 is generally concerned with the rules governing the type and location of pipe breaks required to be postulated. The two documents refer to each other.

Branch Technical Position (BTP) ASB 3-1 Section B.3.a specifies piping failures should be postulated in accordance with BTP MEB 3-1 and that a leakage crack in moderate energy fluid systems piping should be considered separately as a single postulated initial event occurring during normal plant conditions. The ability to mitigate the effects of such piping failure shall consider the most limiting concurrent single active failure. Flooding effects are determined on the basis of a conservatively estimated time period required to effect corrective actions (i.e., detect leak and isolate). The SRP applies the same rules regardless of seismic design .

If the SRP guidance were applied, a moderate energy crack of 1/2 the piping diameter x 1/2 the nominal wall thickness would be assumed at a location which provides the most limiting dilution consequences for adding unborated water to the SFP. The fire protection system for Unit 2 meets the threshold definition of a moderate energy system (defined in Appendix A of Branch Technical Position (BTP) ASB 3-1 as a fluid system that, during normal plant conditions have both a maximum operating temperature of 200 °F or less, and a maximum operating pressure of 275 psig or less). This would be the most limiting case based on line size, inventory available, and driving force. For a nominal 4" schedule 40 fire water line, the postulated crack dimension is 2" x 1/8". Therefore this crack size would be utilized to bound the postulated addition of water to the SFP. Note such evaluation is <u>not</u> required per the current licensing basis.

<u>High Energy Pipe Break/Crack Rules</u>
For the Auxiliary Steam and Condensate Return system, the pipe rupture effects of jet impingement, and the related effects of pressurization, flooding and harsh environmental are required to be addressed. These attributes have previously been addressed in FSAR Section 7.10 by the installation of the auxiliary steam line break detection/isolation (ASDI) system which is designed to rapidly detect and isolate a steam line break or leak, thereby mitigating the potentially adverse effects. Therefore, the potential for the Auxiliary Steam and Condensate Return system line break to result in any significant flooding is precluded by design. The Auxiliary Steam and Condensate Return system meets the high energy line classification (piping systems with normal operating temperature equal to or greater than 200 °F, or normal operating pressure equal to or greater than 275 psig), therefore, a crack in line 6"-HBD-153 is analyzed as the worst case scenario.

5.2 <u>Primary Water</u>

The Primary Water Storage Tank supplies demineralized water to the SFP area. There is a primary water hose station, line 1 ½"-HCD-43, on the 38'-6" elevation north of the cask laydown area (Ref.: 7.3.6). The PWST, which supplies PMW, has a capacity of 150,000 gallons. Makeup to the PWST is a manual evolution and performed by Operations and controlled by procedure. If the contents of the PWST were to be discharged into the SFP, it would be less than the 230,971 gallon dilution limit. This system is designed to seismic Class 2 requirements, but considered a moderate energy line and not postulated to crack under a seismic event. Even if this piping does develop a through wall crack of the size consistent with moderate energy line breaks, the leak flow rate is bounded by the leak rate for the fire protection system, which is 93 gpm. This maximum leak flow rate of 93 gpm is less than the 200 gpm dilution flow rate of unborated water needed to dilute the SFP from 1720 ppm to 600 ppm in 19 hours. Operators would be alerted to this event by a high SFP water level alarm, or PEO rounds identifying a high SFP water level or SFP overflow condition. Therefore, a leak in the Primary Water system is not considered a threat to dilute the SFP below 600 ppm.

5.3 <u>Auxiliary Steam and Condensate Return</u>

The location of the Auxiliary Steam and Condensate Return piping on the 38'-6" elevation northwest and southwest areas of the SFP floor allows for the potential of diluting the SFP. The Auxiliary Steam system is a low pressure steam supply system with a normal operating pressure of 25 psig at 267 °F. The Auxiliary Steam and Condensate Return system meets the high energy line classification (piping systems with normal operating temperature equal to or greater than 200 °F, or normal operating pressure equal to or greater than 275 psig). The worst case scenario would be a break in the 6x4 reducer at the southwest end of the SFP. After the break, steam would emit at sonic velocity as saturated steam, condense, then collect on piping, supports and other structures above the SFP floor. The water volume emitting as a result of this break calculates to 75.4 gpm. This maximum leak flow rate of 75.4 gpm is less than the 200 gpm dilution flow rate needed to dilute the SFP from 1720 ppm to 600 ppm in 19 hours. Operators would be alerted to this event by a high SFP water level alarm, or Plant Equipment Operator rounds identifying a high SFP water level or SFP overflow condition. Therefore, a leak in the Auxiliary Steam and Condensate system is not considered a threat to dilute the SFP below 600 ppm.

5.4 Fire Protection

Fire Protection (FP) System operation is such that a 50 gpm electric jockey pump (M7-11) maintains system pressure by automatically starting when line pressure drops to 105 psig and will run until pressure reaches 120 psig. An electric driven fire pump (P-82) is activated by a single pressure switch set at 85 psig. In the event this switch or pump fails to operate and line pressure continues to drop, the diesel-driven fire pump is activated by an additional pressure switch set at 75 psig. Both the electric and diesel-driven fire pumps deliver 2000 gpm at 100 psi discharge pressure and remain in operation until they are manually shut down. The fire pumps are supplied from two 245,000-gallon ground level suction tanks. The tanks are automatically filled through a water line fed from city water, so there is essentially an unlimited supply of water.

There are two hose stations (HS) on the 38'-6" elevation of the SFP floor that could potentially be a boron dilution path to the SFP, they are HS 230 and HS 226.

The FP system is considered a Moderate Energy Line (MEL) because the system operating conditions are less than 200°F and less than 275 psig. Therefore, piping that meet these operating temperature and pressure limits, requires no postulation of breaks or cracks, based on the original MP2 licensing basis. The FP system HS's on the SFP floor area are being supplied from a 4 inch piping header. To quantify a volume and flow rate from the FP system, the MP2 Hazards Program makes reference to the Standard Review Plan (SRP) NUREG-0800 for guidance in assessing postulated pipe breaks and cracks. In Branch Technical Position ASB 3-1, design breaks or cracks are calculated as ½ the pipe diameter in length and ½ the wall thickness in width. The FP system has been calculated to have a flow rate from a crack to be 93 gpm. This flow rate is less than the 200 gpm of unborated water determined in Section 3.0, which is needed to dilute the SFP boron concentration from 1720 ppm to 600 ppm in 19 hours. Operators would be alerted to this event by a high SFP water level alarm, or PEO rounds identifying a high SFP water level or SFP overflow condition. The minimum time of 19 hours will be more then adequate for operators to notice the alarms, and/or high SFP level condition, locate the source of the leak and isolate the water flow. Therefore, a leak in the Fire Protection system is not considered a threat to dilute the SFP below 600 ppm

5.5 Domestic Water

The Domestic Water system is supplied from the city water supply which has several branch connections in the SFP area. This line, 2"-JDD-10, is considered a moderate energy pipe and therefore, susceptible to the applicable rules for evaluating moderate energy piping for pipe breaks. The Domestic Water system piping and operating parameters are less than the FP system. Even if this piping does develop a through wall crack of the size consistent with moderate energy line breaks, the leak flow rate is bounded by the leak rate for the fire protection system, which is 93 gpm. This maximum leak flow rate of 93 gpm is less than the 200 gpm dilution flow rate of unborated water needed to dilute the SFP from 1720 ppm to 600 ppm in 19 hours. Operators would be alerted to this event by a high SFP water level alarm, or PEO rounds identifying a high SFP water level or SFP overflow condition. Therefore, a leak in the Domestic Water system is not considered a threat to dilute the SFP below 600 ppm.

5.6 Turbine Building Closed Cooling Water

The TBCCW system supplies water to the SFP Cooling Supplemental Cooling Heat Exchanger. The worst case scenario would be a break at the 3"-HBD-434 pipe tee as it enters through the south wall into the SFP area. TBCCW is considered a moderate energy system and therefore, susceptible to the applicable rules for evaluating moderate energy piping for breaks. The TBCCW is a closed loop system with a volume of less than 13,000 gallons. Makeup to the TBCCW system is a manual evolution and performed by Operations and controlled by procedure. The TBCCW system piping and operating parameters are less than the FP system. Even if this piping does develop a through wall crack of the size consistent with moderate energy line breaks, the leak flow rate is bounded by the leak rate for the fire protection system, which is 93 gpm. This maximum leak flow rate of 93 gpm is less than the 200 gpm dilution flow rate of unborated water needed to dilute the SFP from 1720 ppm to 600 ppm in 19 hours. Operators would be alerted to this event by a high SFP water level alarm, or PEO rounds identifying a high SFP water level or SFP overflow condition. Therefore, a leak in the TBCCW system is not considered a threat to dilute the SFP below 600 ppm.

5.7 Roof Drains

The roof drain piping around the SFP is such that the roof drains route from the roof to two separate drain headers.

The 10" roof drain line travels along the south wall of the 38'-6" elevation. This 10" drain line is not seismically supported and any leakage from this line would drain onto the SFP floor. This line interconnects 6 roof drains from the south portion of the auxiliary building roof. Assuming that this entire roof surface area of water drains into the pool, a rainfall of 51 inches would be required in order to challenge the 230,971 gallons dilution limit. Since both a pipe leak and 51 inches of rain would be necessary, this unborated dilution source is not considered a credible threat to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm.

The piping above the SFP is seismically supported and therefore should not be a potential dilution source of water into the SFP. However, should there be a leak in this piping, the roof area supplying this drain piping is less than the non-seismic header. Therefore the required rainfall would be 51 inches or larger. Since both a pipe leak and 51 inches of rain would be necessary, this unborated dilution source is not considered a credible threat to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm.

6.0 CONCLUSIONS

The SFP minimum TS soluble boron limit is proposed to be 1720 ppm. Typical soluble SFP boron concentrations are higher.

Criticality analysis has shown that 600 ppm of soluble boron is needed under normal conditions in the SFP to assure not to exceed the 0.95 k_{eff} design basis (including biases and uncertainties). Further, the criticality analysis has shown that 0 ppm of soluble boron, under normal conditions in the SFP, assures k_{eff} is maintained less than 1.00 (including biases and uncertainties). This engineering analysis of potential scenarios which could dilute the boron concentration in the SFP demonstrates that sufficient time is available to detect and mitigate a boron dilution prior to reaching 600 ppm, thus not exceeding the 0.95 k_{eff} design basis (including biases and uncertainties). It should be noted that for accident conditions, up to 1400 ppm soluble boron is credited in the criticality analysis. However, consideration of a simultaneous occurrence of 2 unlikely and independent events, such as a boron dilution event and another independent accident condition, is not required to be considered by the double contingency principal.

The systems which could dilute the spent fuel pool, either by direct connection to the spent fuel pool, or by a potential pipe crack/break have been analyzed. There is no automatic spent fuel pool level control system in the spent fuel pool, so that any dilution to the spent fuel pool will add water to the spent fuel pool. Therefore, the addition of unborated water to the SFP will lead to increased SFP water level, and if not controlled, an overflow of the SFP.

The ability to prevent the SFP soluble boron concentration from being diluted from the TS minimum value of 1720 ppm to a value of 600 ppm has been demonstrated by showing that each potential dilution source meets one of the following two criteria:

- Any dilution source not capable of supplying 230,971 gallons of unborated water will not be capable of diluting the SFP soluble boron concentration from 1720 ppm to 600 ppm.
- If the dilution flow rate of unborated water is $\leq$ 200 gpm, then at least 19 hours will be needed for the SFP soluble boron concentration to be reduced from 1720 ppm to 600 ppm. All dilution scenarios evaluated here will eventually cause SFP high water level alarms either detected directly by control room alarm, or by the PEO detecting high SFP water levels or SFP overflow. Since the conservatively longest time between PEO rounds is 12 hours, and 19 hours are needed at 200 gpm to dilute the SFP soluble boron concentration to 600 ppm, there is ample time to detect and terminate the dilution event.

The existing potential dilution sources are not credible threats to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm due to either volume or flow rate considerations. The volume of unborated water needed to dilute the SFP soluble boron concentration from 1720 ppm to 600 ppm has been conservatively calculated to be 230,971 gallons. Most of the potential dilution sources described in this report do not have volumes this large, and therefore are not capable of diluting the SFP boron concentration from 1720 ppm to 600 ppm .

For those systems which have the potential to add in excess of 230,971 gallons of water to the SFP, it has been conservatively assumed, in Section 3.0, that a dilution flow rate in excess of 200 gpm of unborated water would be necessary for 19 hours to cause a dilution of the SFP soluble boron concentration from 1720 ppm to 600 ppm. The operators will be alerted to this by high SFP water level alarms in the control room, and/or PEO rounds. Since the conservatively longest time interval between PEO rounds to monitor the SFP general area is 12 hours, and 19 hours are needed at 200 gpm to dilute the SFP soluble boron concentration to 600 ppm, there is ample time to detect and secure the dilution event. Therefore, dilution flow rates less than 200 gpm should be identified either by high SFP water level alarms in the control room, or PEO rounds in sufficient time to detect and isolate a dilution prior to the SFP reaching 600 ppm.

The largest identified dilution flow rate (for any system with a potential dilution volume of at least 230,971 gallons) was 100 gpm from the CST delivered via the AFW system, which is a backup makeup source to the SFP. This value of 100 gpm is far less than the 200 gpm dilution flow rate discussed above. Assuming 200 gpm of unborated CST water is emitted into the SFP, it would take 19 hours (Ref.: Section 3.0) to yield the 230,971 gallons necessary to reduce the SFP boron concentration from 1720 ppm to 600 ppm. This volume is calculated from the TS minimum soluble boron concentration of 1720 ppm to the analyzed 600 ppm required to maintain the SFP less than 0.95 k_{eff}.

If an inadvertent dilution of the SFP occurs using CST water via the auxiliary feed system the operators will be alerted to this by a high SFP water level alarm in the control room, and/or PEO rounds. Since the conservatively longest time interval between PEO rounds to monitor the SFP general area is 12 hours, and 19 hours are needed at 200 gpm (Ref.: Section 3.0) to dilute the SFP soluble boron concentration to 600 ppm, there is ample time to detect and secure the dilution event. Therefore, dilution flow rates less than 200 gpm should be identified either by high SFP water level alarms in the control room, or PEO rounds in sufficient time to detect and isolate a dilution prior to the SFP reaching 600 ppm.

Based on the above evaluation, an unplanned or inadvertent dilution event which would reduce the boron concentration from 1720 ppm to 600 ppm is not credible for MP2. The large volume of water required to dilute the SFP, the TS controls on SFP boron concentration, PEO rounds as well as engineered alarms, would effectively detect a dilution event prior to k_{eff} reaching 0.95. Further, should the SFP boron concentration reach 0 ppm, SFP k_{eff} will still be less than 1.0 (including all biases and uncertainties).