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July 1, 1992

Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: References Requested by June 23, 1992, Memorandum and Order

Dear Atomic Safety and Licensing Board:

In response to the Licensing Board's Memorandum and Order, Licensee Northeast Nuclear Energy Company is providing the following documents:

"Millstone Nuclear Power Station, Unit No. 2 Proposed Revision to Technical Specifications Spent Fuel Pool Reactivity," April 16, 1992; and

Millstone Unit 2 FSAR, Section 14.7.4.3.1 and Table 14.7.4-1.

We have conferred with the NRC Office of General Counsel and understand that they will forward a copy of the Safety Evaluation Report of June 4, 1992 and the May 7, 1992 Supplement to the April 16, 1992 Amendment Application.

Sincerely,

Nicholas S. Reynolds

WINSTON & STRAWN,
ATTORNEYS FOR NORTHEAST NUCLEAR
ENERGY COMPANY

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the Matter of NORTHEAST NUCLEAR ENERGY CO. (Millstone Unit 2)	Docket No. 50-336-OLA

CERTIFICATE OF SERVICE

I hereby certify that copies of:

"Millstone Nuclear Power Station, Unit No. 2 Proposed Revision to Technical Specifications Spent Fuel Pool Reactivity," April 16, 1992; and

Millstone Unit 2 FSAR, section 14.7.4.3.1 and Table 14.7.4-1

have been served by U.S. Mail, first class, on this 1st day of July, 1992, as follows:

Office of Commission Appellate
Adjudication
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Administrative Judge
Ivan W. Smith, Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Administrative Judge
Charles N. Kelber
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Administrative Judge
Jerry R. Kline
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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NORTHEAST NUCLEAR ENERGY CO.

Nicholas S. Reynolds

WINSTON & STRAWN,
ATTORNEYS FOR NORTHEAST NUCLEAR
ENERGY CO.

July 1, 1992

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

April 16, 1992

Docket No. 50-336
B14102
Re: 10CFR50.90
10CFR50.91

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

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Spent Fuel Pool Reactivity

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend its Operating License DPR-65, by incorporating the changes identified in Attachment 1 into the technical specifications of Millstone Unit No. 2.

Description of the Proposed Changes

The proposed change to the Millstone Unit No. 2 Technical Specifications would modify the existing two region spent fuel pool design, modified by amendment 109, dated January 15, 1986[1] and amendment 128, dated March 31, 1988,[2] to a three region configuration.

Presently, Region I is designed to store up to 384 fuel assemblies with an initial enrichment of up to 4.5 weight percent U-235. Region I is comprised of 5 rack modules and fuel assemblies can be stored in every location. The Region I racks contain a neutron poison material (Boraflex), and have a nominal center-to-center distance between storage locations of 9.8 inches. Region II is designed to store up to 728 fuel assemblies which have sustained their design burnup. Fuel assemblies are stored in a three out of four array, with blocking devices installed to prevent inadvertent placement or storage of a fuel assembly in the fourth location. The Region II storage racks have a nominal center-to-center distance between storage locations of 9 inches.

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(1) D. B. Osborne letter to J. F. Opeka, [Issuance of Amendment 109], dated January 15, 1986.

(2) D. H. Jaffe letter to E. J. Mroczka, "Issuance of Amendment (TAC No. 65274)," dated March 31, 1988.

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OS3422 REV 4-88

The proposed changes will result in a three region configuration, which will be described by alphabetic letters rather than the previous numeric convention. Region A will utilize three of the existing Region I poison rack modules. Region A is designed to store up to 224 fuel assemblies, which will be qualified for storage in this region by verification of adequate assembly average burnup versus fuel assembly initial enrichment (reactivity credit for burnup). Fuel assemblies can be stored in every location in Region A. These racks will be used for immediate storage of fuel discharged from the reactor. Region B will utilize the remaining two existing Region I poison rack modules. Region B is designed to store up to 120 new fuel assemblies with an initial enrichment of up to 4.5 weight percent U-235 and other assemblies which do not satisfy the burnup versus initial enrichment requirements of either Region A or Region C. Fuel assemblies will be stored in a three out of four array in Region B, with blocking devices installed to prevent inadvertent placement or storage of a fuel assembly in the fourth location. Region C is the new designation for the existing Region II storage racks. This alphabetic storage rack designation is a human factors consideration, designed to minimize the probability of a fuel assembly movement error and to provide a historical distinction between the various fuel pool configuration records.

The following details the proposed changes to the Technical Specifications:

1) Definition 1.39, STORAGE PATTERN is currently defined for Region II. This is being changed to define the 3-out-of-4 array to be used in Regions B and C.

2) Specification 3.9.17 is currently concerned with fuel movement over Region II racks (due to the dropped assembly accident and misplaced fuel assembly event). This is being changed from any fuel movement over the Region II racks to any fuel movement in the spent fuel pool.

3) Specification 3.9.18 is being modified to change the wording in the surveillance requirements from Region II to Region C, and adds a surveillance requirement to ensure that fuel assemblies to be placed in Region A are within the enrichment and burnup limits of a new Figure (3.9-4).

4) Figure 3.9-1 is being modified to change the references from Region II to Region C.

5) Figure 3.9-2 is being modified to delete the references from Regions I and II and add Regions A, B and C.

6) Figure 3.9-3 is being modified to change the references from Region II to Region C.

7) A new Figure (3.9-4) is being added to specify the allowable enrichment and burnup limits for fuel assemblies to be stored in Region A.

8) Specification 3.9.19 is being split into two parts:

   (a) Specification 3.9.19.1 is the old specification 3.9.19, changing the references from Region II to Region C.

   (b) Specification 3.9.19.2 is a new requirement for the STORAGE PATTERN requirements of Region B.

9) The Design Features section for Fuel Storage Criticality and Capacity are being changed to describe the design features for the newly defined regions (A, B, and C), as well as to change the storage capacity numbers to reflect the blocked locations in Regions B and C.

10) The Bases sections for Specifications 3.9.17, 3.9.18, and 3.9.19 are being changed to reflect the changes introduced by the changes in the spent fuel storage rack criticality design basis.

The proposed revisions to Sections 1, 3/4 9, 5, and Bases associated with this amendment are included in Attachment 1.

Reason for the Change

These changes to the Millstone Unit No. 2 Technical Specifications are being proposed as a result of the errors recently discovered in the spent fuel rack criticality analysis. This information was shared with NRC Staff personnel, in a timely manner, via a prompt report on February 14, 1992 in accordance with 10CFR50.72 and several follow-up telephone conversations. Licensee Event Report 92-003-00, dated March 13, 1992,(3) reported this in accordance with 10CFR50.73(a)(2)(ii)(B). The calculational errors were discovered while performing criticality reanalyses associated with the Boraflex degradation. These proposed changes, as well as design modifications to the spent fuel pool storage racks (addition of cell blocking devices) are required to provide fuel storage for the upcoming refueling outage.

Safety Assessment

The safety assessments associated with these changes have considered the mechanical, material, thermal, seismic/structural, and reactivity (potential criticality) aspects of the spent fuel pool. All previously evaluated accidents associated with the spent fuel racks are also addressed. The assessment considers the functional design aspects of the fuel rack cell blocking devices as they relate to the spent fuel racks and accident conditions. Spent fuel pool criticality safety analyses are included as Attachment 2.

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(3) S. E. Scace letter to U.S. Nuclear Regulatory Commission, Facility Operating License No. DPR-65, Docket No. 50-336 "LER 92-003-00," dated March 13, 1992.

In accordance with 10CFR50.92, NNECO has reviewed the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed change does not involve a significant hazards consideration because the change would not:

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

   Radiological consequences of the fuel handling accident are not impacted by the formation of Regions A and B because the fuel assembly design is unchanged. However, the probability of occurrence of a fuel misplacement error has increased slightly. The increase is not significant because the types of controls being put into place in Regions A and B are of the same type as already in place in Region C. Furthermore, a fuel assembly misplacement error is not considered an accident, as defined in the Final Safety Analysis Report.

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

   No changes are being made to the fuel assemblies or the storage racks, and controls used in the fuel pool will be of the same type as are now in place. As such, there is no possibility of a new or different kind of accident being created. The existing design basis covers all possible accident scenarios in the spent fuel pool.

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

   There is no reduction in the margin of safety since Keff $\leq$ 0.95 is met under all analyzed conditions using conservative assumptions which do not credit the soluble boron in the spent fuel pool except under some accident conditions, as allowed by NRC guidelines. The original mechanical analyses are unchanged for thermal and seismic/structural considerations.

Moreover, the Commission has provided guidance concerning the application of the standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51FR7751) of amendments that are considered not likely to involve a significant hazards consideration. The proposed change is similar to example (ii) which is a change that constitutes an additional limitation, restriction, or control not presently included in the technical specification. The definition of an additional spent fuel pool storage area (with additional administrative controls) for fuel assemblies constitutes an additional limitation not presently included in technical specifications. The consequences remain unchanged and the margin of safety is not impacted.

Based on the information contained in this submittal and the environmental assessment for Millstone Unit No. 2, there are no significant radiological or nonradiological impacts associated with the proposed action, and the proposed license amendment will not have a significant effect on the quality of the human environment.

The Millstone Unit No. 2 Plant Operations Review Committee has reviewed and recommended approval of the proposed license amendment. The Nuclear Review Board has reviewed this proposed license amendment and has concurred with the above determination.

Regarding our proposed schedule for this amendment, we request issuance at your earliest convenience, considering our current refuel outage start date of May 30, 1992, with amendment effective as of the date of issuance, to be implemented within 30 days of issuance.

In accordance with 10CFR50.91(b), we are hereby providing the State of Connecticut with a copy of this proposed amendment.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Executive Vice President

cc: T. T. Martin, Region I Administrator
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Mr. Kevin McCarthy
Director, Radiation Control Unit
Department of Environmental Protection
Hartford, Connecticut 06116

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me, J. F. Opeka, who being duly sworn, did state that he is Executive Vice President of Northeast Nuclear Energy Company, a Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensee herein, and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. D'Amico
Notary Public

My Commission Expires March 31, 1993

Docket No. 50-336
B14102

Attachment 1

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Spent Fuel Pool Reactivity

Proposed Revised Pages

9204270157

25 pp

April 1992

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.9 REFUELING OPERATIONS

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

VENTING

1.35 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a VENTING process.

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

The term "REAL MEMBER OF THE PUBLIC" means an individual who is exposed to existing dose pathways at one particular location.

SITE BOUNDARY

1.37 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial institutional and/or recreational purposes.

STORAGE PATTERN

1.39 The Region B and C spent fuel racks contain a cell blocking device in every 4th rack location for administrative control. This 4th location will be referred to as the blocked location. A STORAGE PATTERN refers to a blocked location and all adjacent and diagonal cell locations surrounding the blocked location within the respective region.

MOVEMENT OF FUEL IN SPENT FUEL POOL

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 maintained uniform and sufficient to maintain a boron concentration of 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 REQUIREMENT

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.

SPENT FUEL POOL--REACTIVITY CONDITION

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 REQUIREMENT

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 are within the enrichment and burn-up limits of Figure 3.9.1 by checking the assembly's design and burn-up documentation.

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.

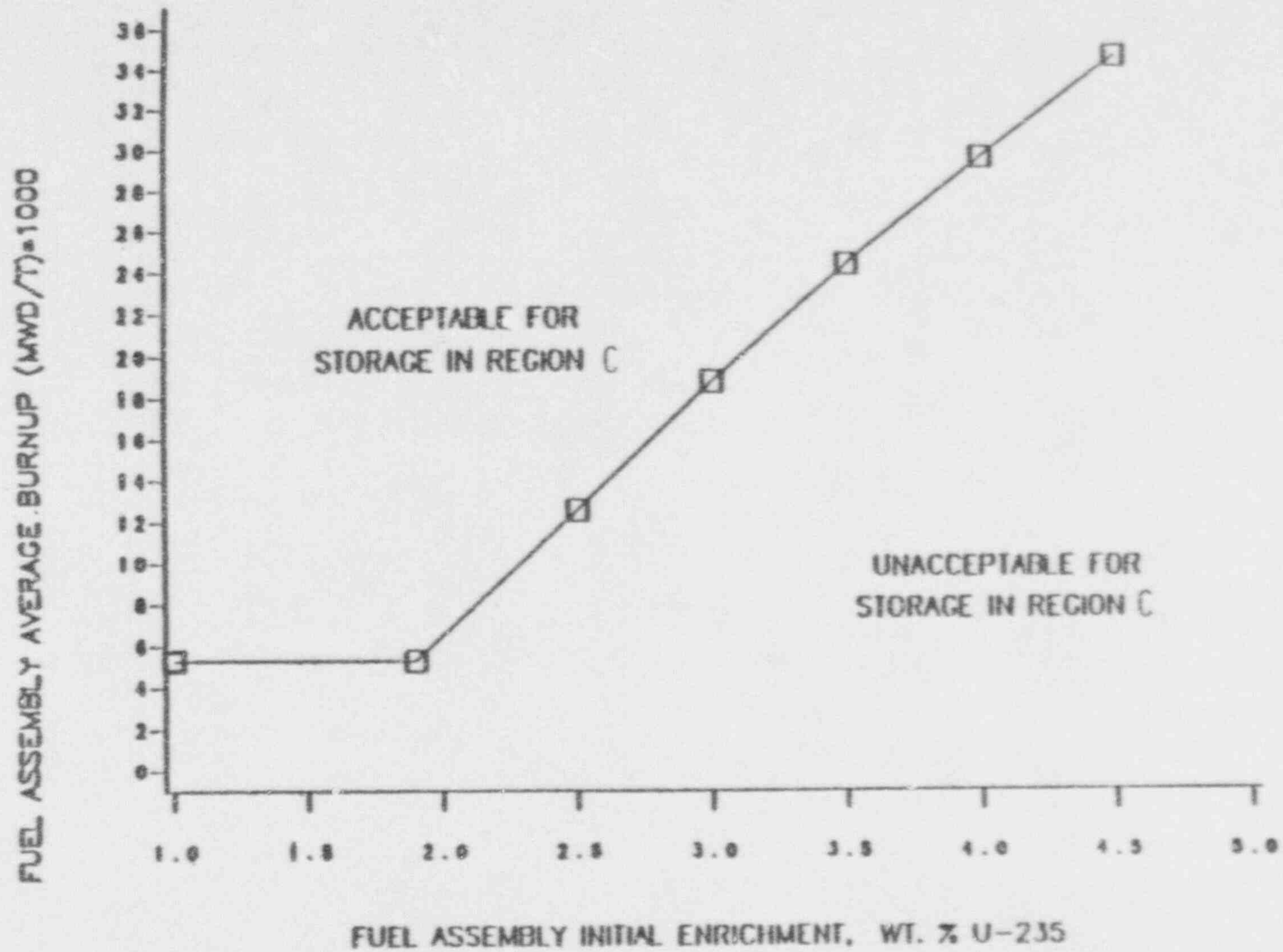
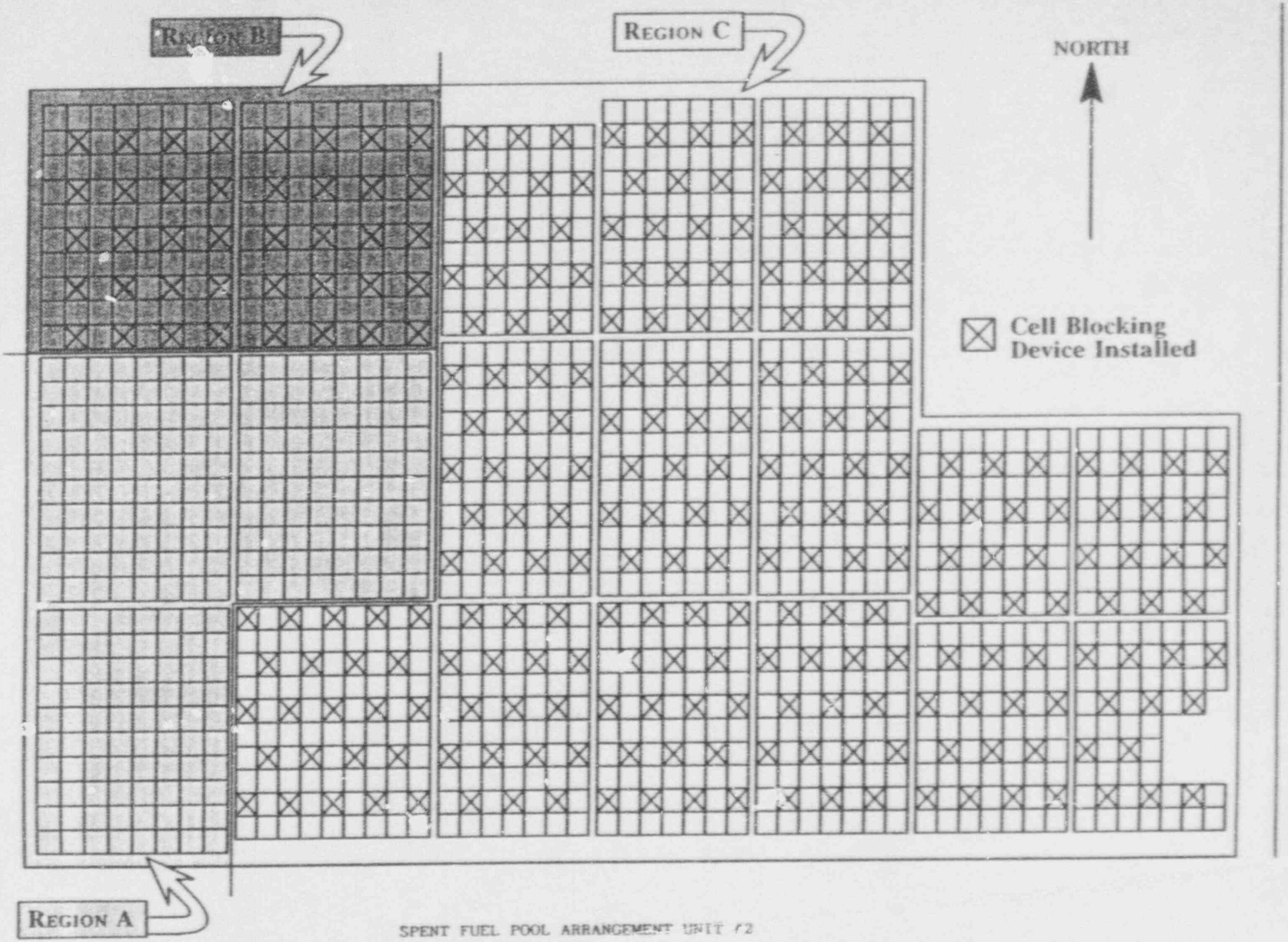


FIGURE 3.9-1 MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION C



SPENT FUEL POOL ARRANGEMENT UNIT #2

FIGURE 3.9-2

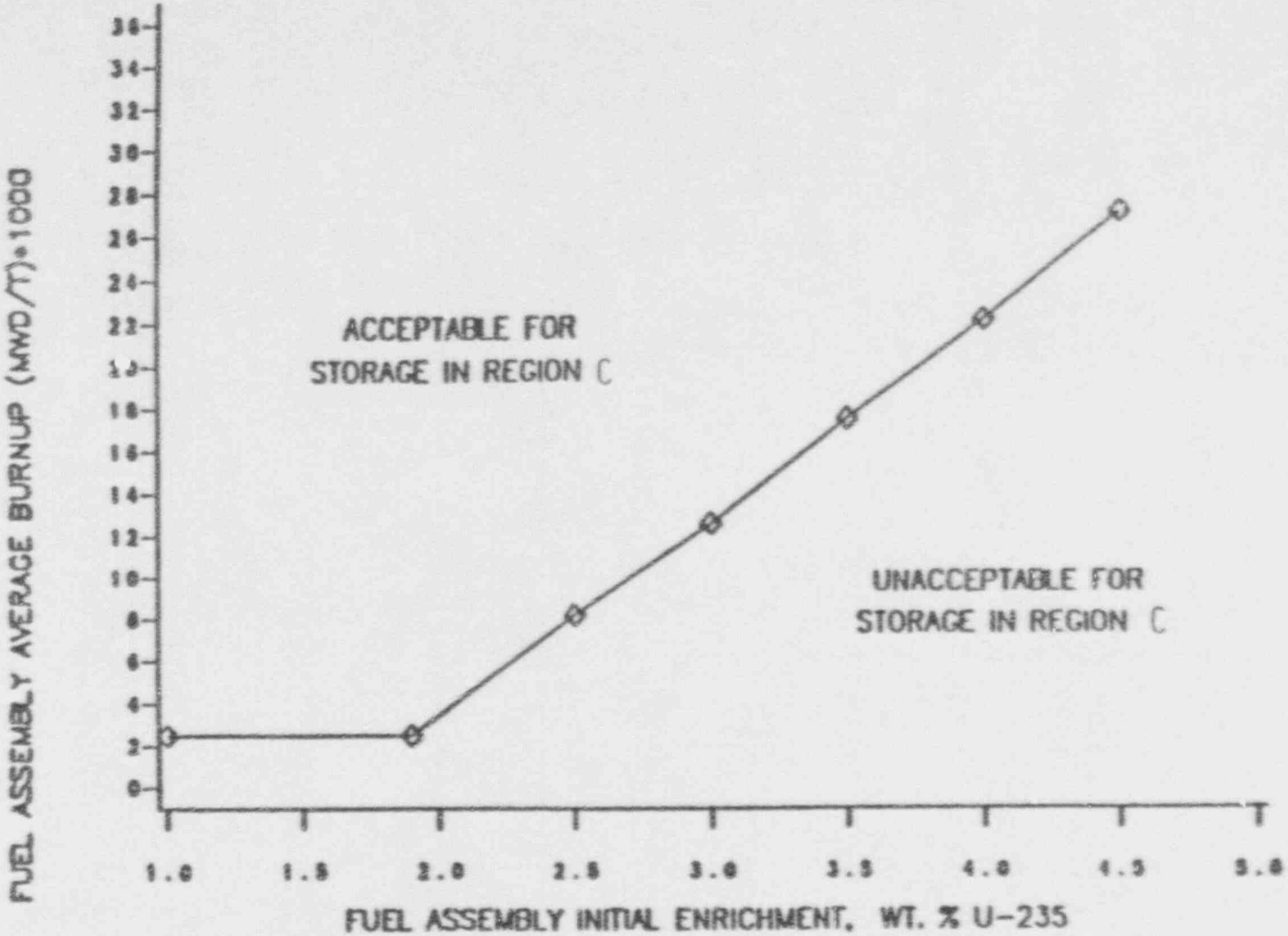


FIGURE 3.9-3 MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION C AS CONSOLIDATED FUEL

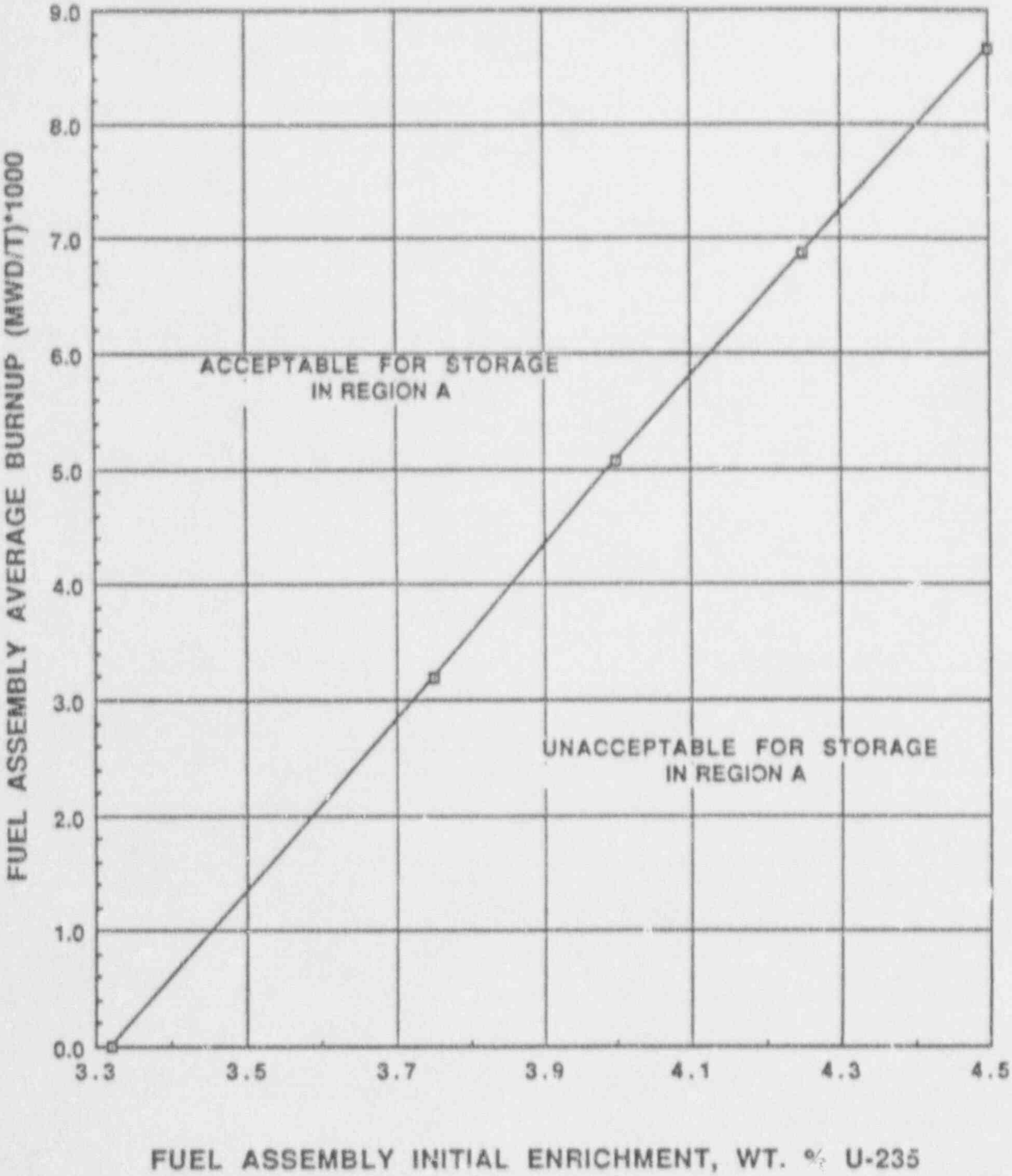


FIG. 3.9-4 MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION A

SPENT FUEL POOL - STORAGE PATTERN

LIMITING CONDITION FOR OPERATION

3.9.19.1 Each STORAGE PATTERN of the Region C spent fuel pool racks shall require either 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 of the STORAGE PATTERN must have consolidated fuel in them, including the formerly blocked location; or

(3) Meet both (a) and (b):

(a) If a cell blocking device has been removed, all cells of the STORAGE PATTERN must have consolidated fuel in them except the formerly blocked location.

(b) The formerly blocked location is vacant and a consolidated fuel box or cell blocking device is immediately being placed into the formerly blocked cell.

APPLICABILITY: Fuel in the Spent Fuel Pool

ACTION:

Take immediate action to comply with either 3.9.19.1(1), (2) or (3).

SURVEILLANCE REQUIREMENTS

4.9.19.1 Verify that 3.9.19.1 is satisfied at the following times.

(1) Prior to removing a cell blocking device

(2) Prior to removing a consolidated fuel storage box from its Region C storage location.

SPENT FUEL POOL - STORAGE PATTERN

LIMITING CONDITION FOR OPERATION

3.9.19.2 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.

APPLICABILITY: Fuel in the spent fuel pool.

ACTION:

Take immediate action to comply with either 3.9.19.2(1) or (2).

SURVEILLANCE REQUIREMENTS

4.9.19.2 Verify that 3.9.19.2 is satisfied prior to removing a cell blocking device.

3/4.9.13 STORAGE POOL RADIATION MONITORING

The OPERABILITY of the storage pool radiation monitors ensures that sufficient radiation monitoring capability is available to detect excessive radiation levels resulting from 1) the inadvertent lowering of the storage pool water level or 2) the release of activity from an irradiated fuel assembly.

3/4.9.14 & 3/4.9.15 STORAGE POOL AREA VENTILATION SYSTEM

The limitations on the storage pool area ventilation system ensures that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.16 SHIELDED CASK

The limitations of this specification ensure that in an event of a cask tilt accident 1) the doses from ruptured fuel assemblies will be within the assumptions of the safety analyses, 2) K_{eff} will remain $\leq$ .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 assembly or a consolidated fuel storage box drop accident into a Region B or C rack location completing a 4-out-of-4 fuel assembly geometry, K_{eff} will remain $\leq$ 0.95.

3/4.9.18 SPENT FUEL POOL - REACTIVITY CONDITION

The limitations described by Figures 3.9-1 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, introducted 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 conditions of the Region B and C 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 C storage racks are designed to prevent inadvertent placement and/or storage of fuel assemblies 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. Only loaded consolidated fuel storage boxes may be placed and/or stored in the 4th location, completing the STORAGE PATTERN, after all adjacent, and diagonal, locations are occupied by loaded consolidated fuel storage boxes.

The Cell Blocking Devices is 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.

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.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,060 + 700/-0 cubic feet.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original 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 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 Figure 3.9-1 to ensure that the design burn-up has been sustained. Fuel assemblies stored in this region are placed in a 3 out of 4 STORAGE PATTERN for reactivity control. The contents of consolidated fuel storage boxes to be stored in this region must comply with Figure 3.9-3.

e) Region C of the spent fuel storage pool is designed to permit storage of consolidated fuel in the 4th location of the storage rack and ensure a $K_{eff} \leq 0.95$. Placement of consolidated fuel in the 4th location is only permitted if all surrounding cells of the STORAGE PATTERN are occupied by consolidated fuel.

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 1237 storage locations to receive spent fuel and 109 storage locations to remain blocked.

<u>Docket No. 50-336</u>
<u>B14102</u>

Attachment 2

'illstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Spent Fuel Pool Reactivity

Spent Fuel Pool Criticality Safety Analyses

April 1992

Spent Fuel Pool Criticality Safety Analyses

This attachment is intended to document results of our criticality safety analyses of the Millstone Unit No. 2 Region 1 storage cells with observed and postulated gaps present in the Boraflex absorber material. The Boraflex poison degradation has been very conservatively incorporated into the criticality design analysis. To date, approximately half of the poisoned rack cells have been tested and characterized for gap formations. Test data identifies a Boraflex panel defect rate of 16% with the largest observed gaps at a 2% shrinkage rate. With further gap growth anticipated, the mechanical inputs for the criticality analysis assumed 4% gap formations at the observed test locations and a 4% gap formation with a random distribution in <u>all</u> of the other Boraflex panels. These assumptions are considered conservative because EPRI data supports the 4% maximum shrinkage value and the random distribution is supported by the NNECO test data. These analyses are based on the CE design for the Region 1 Boraflex poisoned racks, as originally licensed for Millstone Unit No. 2 in Amendment #109, dated January 15, 1986. The calculations utilize a three-dimensional NITAWL-KENO-5a model with the 27-group SCALE cross-section set.

Sections of the old Region 1 have been redefined as two new regions:

Region A utilizing all of the cells in a 4-of-4 cell arrangement with credit for fuel burnup.

Region B using fresh fuel of 4.5% average enrichment in a 3-of-4 arrangement (fourth cell empty).

Shrinkage of 4% was also assumed resulting in 5.65" gaps in every Boraflex panel (a very conservative assumption). The consequence of various axial distributions was also investigated. A shrinkage of 4% in width was conservatively assumed although examination of the Boraflex from Cell D9 did not show any visible evidence of such shrinkage.

Table 1 summarizes results of several calculations (including the original design with fresh fuel in every location) intended to show the magnitude of the reactivity effects of in the Millstone Unit No. 2 racks.

To provide some perspective for the analyses, a calculation was made assuming all the Boraflex was lost, resulting in a 0.194 δk total reactivity "worth" of the Boraflex. If 4% is lost through gap formation, then the order of magnitude of the expected reactivity effect due to gaps is $(0.04 * 0.194) = 0.008\ \delta k$.

Region B: 3 of 4 Cell Arrangement with 4.5% Fresh Fuel

Calculations for a 3-of-4 arrangement with fresh 4.5% enriched fuel (fourth cell empty) are summarized in Table 2. At the present time, the gaps which have been observed in the Boraflex (as of the most recent Blackness test) have a negligible reactivity effect within the statistical accuracy of KENO-5a calculations.

Considering that the Boraflex has already seen three fuel cycles, it is not likely that significant further growth would be expected. However, for conservatism, it was assumed that these gaps increase to 5.65" (equivalent to 4% axial shrinkage) at the locations observed in the Blackness tests (Case 7, Table 2).

To assure very conservative upper bound conditions, further calculations were made assuming that additional gaps of 5.65" appear in all other panels throughout the racks. Based on the fact that the axial distribution of observed gaps is random, a random distribution of these additional gaps in the axial direction was assumed as the reference case. (Gap locations were derived by using a PC random number generator.) The maximum k_{eff} for the upper bound reference case (Case 9, Table 2) was calculated to be 0.9179, including width shrinkage bias and all uncertainties (calculational and manufacturing tolerances, see Table 3). Thus, with the 3-of-4 arrangement, the maximum k_{eff} remains substantially below the NRC criterion (0.95 k_{eff}).

Westinghouse and CE fuel show a slightly higher reactivity than the ANF fuel used for the primary analyses. For Westinghouse and CE fuel, the maximum reactivities for 4.5% enriched fuel were calculated to be 0.9252 and 0.9201 respectively.

The temperature and void coefficients of reactivity are negative. Therefore, the calculations were conservatively based on a temperature of 4°C (maximum water density) and any temperature increase above 4°C would result in reduced reactivity.

Two accident conditions were also considered, as follows: (Note: Under the accepted single failure criterion, it is not necessary to consider the simultaneous occurrence of multiple independent accident conditions. Therefore, credit for the presence of soluble poison is allowed under accident conditions.)

- Mislocated fuel assembly--For the case of a fresh fuel assembly assumed to be accidentally installed into one of the empty cells of an otherwise filled Region B array, the maximum k_{eff} was calculated to be 0.9436, which remains below the NRC criterion.

- Mislocated Consolidated fuel assembly--This accident assumes that a consolidated fuel bundle is accidentally loaded into one of the empty cells of Region B. Calculations for this case resulted in a maximum k_{eff} of 0.9364 which is well within the NRC criterion. This is a very conservative calculation that assumes a consolidation ratio of 2 with unburned rods of 4.5% enrichment rather than spent fuel rods.

Significance of the Axial Gap Distribution

Since the potential effect of the axial gap distribution is of concern, we have calculated the reactivity effect of 5.65" gaps for several assumed distributions. The actual distribution of gaps in the Millstone Unit No. 2 Boraflex appears to be random or very nearly so. Blackness Tests conducted in many plants generally substantiate the assumption of a random distribution.

Calculations for two assumed distributions are summarized in Table 4. On the basis of this evaluation, it is concluded that the distribution of gaps in the axial direction has a comparatively minor impact on the distribution. The observed gap distribution (augmented to 5.65" at all gap locations plus a random distribution of 5.65" gaps in Boraflex panels which did not have gaps) yields the same reactivity as the assumption of a completely random distribution of the same size gaps. For the extreme (and noncredible) case of all gaps assumed to occur only in the central 50% of the rack height, the k_{eff} was 0.005 δk above the randomly distributed case, and for an assumed cosine distribution of gaps, the reactivity was 0.0028 δk higher than the reference random distribution. Neither of these hypothetical distributions would result in exceeding the NRC criterion.

In addition, we have investigated the consequences of the Boraflex shrinkage resulting in a reduction in length of 5.65" (4%) exposing an unpoisoned zone at the ends of the fuel assemblies. This case resulted in a smaller reactivity effect than the case of gaps distributed throughout the rack.

Region A: 4-of-4 Cell Arrangement with Burnup Credit

The storage racks are capable of accepting spent fuel utilizing all cells. Calculations have been made for the storage racks loaded in a 4-of-4 arrangement with spent fuel of a specified minimum burnup. Since the required burnup is not large, we selected a conservative value for the design k_{eff} (0.9317 with fuel of 4.5% enrichment and 8670 MWD/MTU burnup) knowing that most, if not all, of the spent fuel have burnups well in excess of the minimum required.

Thus, a conservative value may be used without significant impact on Millstone Unit No. 2 operations. With this design basis reactivity, the misloading of either a fresh fuel assembly or a consolidated fuel bundle will not result in exceeding the NRC criterion.

Table 3 summarizes the uncertainties for Region A, based upon fuel of 4.5% initial enrichment burned to 8670 MWD/MTU. With these uncertainties, the maximum k_{eff} is 0.9317 (95% probability at the 95% confidence level). Calculations were also made for other assumed initial enrichments and a curve of limiting burnup (for the same reactivity) is presented in Figure 3.9-4 of Technical Specifications. With Westinghouse or CE fuel of 4.5% initial average enrichment, the burnup limit curve will be the same although the calculated reactivities will be slightly higher (0.9381 and 0.9335 for Westinghouse and CE fuel respectively). Discharged fuel would normally be expected to have burnups considerably in excess of the minimum required, resulting in a much lower reactivity.

Calculations for Region A were also made to determine the effect of the axial distribution in burnup. At the low design basis burnup for Region A, no effect was expected and calculations showed that the k_{eff} with axially distributed burnups is less than that of the reference uniform burnup case. (See also Turner, "An Uncertainty Analysis--Axial Burnup Distribution Effects" in Sandia Report SAND89-018, October 1989.)

<u>Interfaces with Other Regions</u>

Calculations were also made to determine if there might be any adverse reactivity effects along the interface between regions. Even without credit for the isolating water-gap between modules, no adverse effects were found for any of the interfaces--Regions A and B, Regions A and C, and Regions B and C (see Figure 3.9-2 in Technical Specifications). Region C is the old Region II, designed for burned fuel.

Based upon the analyses performed, it is concluded that, in the presence of the conservatively postulated maximum gaps (4% or 5.65") in all Boraflex panels, 4% shrinkage in width, and all uncertainties included, that

(1) the Millstone Unit No. 2 spent fuel storage racks can safely accommodate fresh 4.5% enriched fuel in a 3 out of 4 loading pattern with the fourth cell empty.

(2) the Millstone Unit No. 2 spent fuel storage racks can safely accommodate spent fuel of the burnup-enrichment combinations indicated in Figure 3.9-4 of the Technical Specifications, using all cells in a 4 out of 4 arrangement.

(3) no credible accident condition will result in exceeding the regulatory reactivity limit of $k_{eff} \leq .95$.

(4) the assumed axial gap distribution has only a minor impact on the calculated reactivity of the racks (measured distribution used in the reference case analysis), and, for any credible assumption of the distribution of postulated gaps, the maximum k_{eff} will remain within NRC criterion.

Table 1

Background KENO-5a Calculations
(width shrinkage not included)

No.	Case	Max k_{eff}
1	Original Design (4 of 4), no gaps	0.9812
2	4 of 4 Loading, random 5.65" gaps	0.9879
3	3 of 4 Loading, no gaps	0.9113
4	3 of 4 loading, random 5.65" gaps	0.9163
5	no Boraflex	1.0838

Table 2

Criticality Calculations for 3 of 4 Loading Arrangement
(4.5% Enriched Fuel - No Burnup)

No.	Case	Max k_{eff}
3	3 of 4 loading pattern, no gaps	0.9113
6	With gaps as measured in Blackness Testing	0.9110
7	Observed Gaps increased to 5.65"	0.9126
8	Observed Gaps increased to 5.65" plus 5.65" gaps randomly distributed in all other Boraflex Panels	0.9163
9*	Same as Case 8 but with 4% shrinkage in width of the Boraflex	0.9179*
10	Reference Case (9) with Westinghouse fuel	0.9251
11	Reference Case (9) with CE fuel	0.9201
12	Accident of a Fresh 4.5% assembly installed in an empty cell	0.9420
13	Accident of 4.5% Consolidated Bundle installed in an empty cell	0.9348

* Reference Case

Table 3

Calculations Uncertainties and Reactivity Effects of Manufacturing Tolerances

Item	Reactivity δk Region A	Reactivity δk Region B
Uncertainty in Bias	± 0.0018	
KENO Statistics (95%/95%)	± 0.0019[4] (or ± 0.0012[5])	
B-10 Loading Tolerance (± 0.003 g B-10/cm²)	± 0.0022	± 0.0020
Boraflex Width (± 1/16")	± 0.0009	± 0.0016
Enrichment Tolerance (± 0.09%)	± 0.0020	± 0.0020
UO_2 Density Tolerance (± 2%)	± 0.0021	± 0.0021
Lattice Spacing (± 0.09")	± 0.0096	± 0.0113
SS Box ID (± 0.05")	± 0.0042	± 0.0073
SS wall thickness (± 0.012")	± 0.0015	± 0.0053
Uncertainty in Depletion Calculations (5% in burnup)	NA	± 0.0028
Statistical Average	± 0.0115 (or ± 0.0114)	± 0.0154 (or ± 0.0153)

(4) For 1000 generations of 500 neutrons each.

(5) For 2500 generations of 500 neutrons each.

Table 4

Significance of the Axial Distribution of Gaps Locations
(width shrinkage of the Boraflex not included)

No.	Case	Max k_{eff}
3	3 of 4 arrangement, no gaps	0.9113
8	With observed gap locations (5.65" gaps) and a random distribution of 5.65" gaps in all other Boraflex panels	0.9163
9A	With a random distribution of 5.65" gaps in all panels	0.9163
15	With an assumed cosine distribution in gap locations	0.9191
16	Random distribution to gap locations in the central 50% of the axial height	0.9212

TABLE 14.7.4-1

Assumption for Fuel Handling Accident in the Spent Fuel Pool

Assumption	Basis
(1) Reactor Core Power Level = 2700 MWt	Stretch Power
(2) Iodine Pool Decontamination Factor = 100	Reg. Guide 1.25
(3) Activity Released from Rods: Iodines = 10% Noble Gases (Except KR-85) = 10% KR-85 = 30%	Reg. Guide 1.25
(4) Chemical Form of Iodines Above the Pool: 25% in Organic Form 75% in Inorganic Form	Reg. Guide 1.25
(5) a) One Assembly Assumed to Rupture b) 14 Rods Assumed to Rupture	a) Reg. Guide 1.25 b) FSAR
(6) Decay Time = 72 Hours 120 Days 5 Years	Tech. Spec. #3.9.3 Tech. Spec. #3.9.16.1 Tech. Spec. #3.9.20
(7) Number of Assemblies in Core = 217	FSAR
(8) EBFS Filter Efficiencies: Organic Iodine = 70% Elemental Iodine = 90%	Reg. Guide 1.25/MP2 SER
(9) All Activity Released from Fuel Pool Building Instantaneously Through Filters	Reg. Guide 1.25
(10) X/Qs (sec/m^3) (for MP1 Stack Release) Site Boundary (0-1 Hour) = (1.03E-4) LPZ (0-1 Hour) = 3.41E-5	95% Maximum X/Qs during the years 1974-1976
(11) Thyroid Dose Conversion Factors from Reg. Guide 1.109	See Justification Under Section V, LOCA
(12) Semi-Infinite Cloud Dose Model	Reg. Guide 1.25
(13) Peaking Factor = 1.65	Reg. Guide 1.25
(14) Breathing Rate = 3.97×10^{-4} m^3/sec.	Reg. Guide 1.25

Release of activity through the containment purge system would be prevented by automatic closure of the containment isolation dampers as described in Subsection 9.9.2.2. The containment personnel hatches and equipment hatches are closed during fuel handling operations.

Since the auxiliary building cannot be completely isolated, this results in a more limiting activity release to the environment. Prior to the handling of irradiated fuel, the exhaust air is diverted from the main exhaust system by being manually aligned to the auxiliary exhaust system (AES) and exhausted from the spent fuel pool area through the enclosure building filtration system (EBFS) charcoal filter to remove iodines (see Subsection 9.9.8) prior to release through the Unit 1 stack.

14.7.4.2 Method of Analysis

For the purpose of defining the upper limit on fuel damage as the result of a fuel handling accident, it is assumed that the fuel assembly or consolidated fuel storage box is dropped during handling. Interlocks, procedural and administrative controls make such an event unlikely. However, if an assembly is damaged to the extent that a number of fuel rods fail, the accumulated fission gases and iodines in the fuel element gap could be released to the surrounding water. Release of the fission products to the surrounding water is considered negligible as a result of reduced diffusion through the fuel due to the low fuel temperature during refueling.

The fuel assemblies and consolidated fuel storage box are stored within the spent fuel rack at the bottom of the spent fuel pool. The top of the rack extends above the top of the stored fuel. A dropped fuel assembly or consolidated fuel storage box could not strike more than one fuel assembly in the storage rack. Impact can occur only between the ends of the involved components, the bottom end fitting of the dropped components impacting against the top end fitting of the stored fuel assembly. The results of an analysis on the energy absorption capability of a fuel assembly indicate that a fuel assembly is capable of absorbing the kinetic energy of the fuel assembly or consolidated fuel storage box drop with no fuel rod failures. The worst fuel handling incident that could occur in the spent fuel pool is the dropping of a fuel assembly to the fuel pool floor. The dropping of a consolidated fuel storage box was evaluated and determined to be bounded by the fuel assembly drop to the fuel pool floor. After striking the pool floor vertically, the assembly would rotate into a horizontal attitude. It is postulated that during this rotation the assembly will strike a protruding structure. The fuel storage pool has been designed without such a protruding structure, hence, the shape and nature of the assumed member is indeterminate. For this analysis, therefore, a line load has been assumed.

To obtain an estimate of the number of fuel rods which might fail in the event a fuel assembly is dropped, the energy required to crush a fuel rod and bend the entire assembly has been determined. The point of impact was assumed to be the most effective location for fuel rod damage, the center of percussion. Resistance to crushing offered by the fuel pellet is considered in the analysis. Failure of the fuel tube by crushing absorbs the least energy, hence, the model produces a conservative upper limit for the number of fuel rod failures. This failure mode is applicable to the outer row of fuel rods only. Since it is not possible to apply a line load beyond the outer row of fuel rods, the failure mode of rods in rows other than the outer rows will be by bending rather than by crushing.

Approximately 36,000 in.-lbs of kinetic energy from rotation must be absorbed. The energy required to bend the assembly and crush the outer row of fuel rods to failure is 4,600 in.-lb. Failure of the second row of fuel rods by bending along requires more than 70,000 in.-lbs. Thus, no more than 14 fuel rods, i.e., one outer row of rods, would be expected to fail.

All X/Q values have been chosen in the following manner: Site meteorological data has been examined for the years 1974, 1975, and 1976. For each release point and dose calculation time period in question, the year with the largest (most conservative) 95% maximum X/Q value has been chosen.

For each accident, the results indicate that for operation of Millstone Unit No. 2 at 2700 MWt, the radiological consequences will not exceed the limitations of 10CFR100, and are in fact significantly below the limits in most cases.

14.7.4.2.1 Fuel Handling Accident in the Spent Fuel Pool

This accident has been reanalyzed using the assumptions contained in Regulatory Guide 1.25. A complete list of assumptions is provided in Table 14.7.4-1. The results of this analysis, which are well below the limits of 10CFR100, are summarized in Section 14.7.4.3.1.

14.7.4.2.2 Fuel Handling Accident in Containment

A complete list of the assumptions used in this calculation is provided in Table 14.7.4-2. The results of the analysis, which are well within the limits of 10CFR100, are summarized in Section 14.7.4.3.2.

14.7.4.3 Results of Analysis

14.7.4.3.1 Spent Fuel Pool Accident

	Dose (rems)			
	Site Boundary		LPZ	
Organ	One Assembly	14 Rods	One Assembly	14 Rods
Thyroid	3.3	2.7×10^{-1}	1.1	8.9×10^{-2}
Whole Body	7.9×10^{-2}	6.4×10^{-3}	2.6×10^{-2}	2.1×10^{-3}

14.7.4.3.2 Containment Accident

	Dose (rems)			
	Site Boundary		LPZ	
Organ	14 Rods	One Assembly	14 Rods	One Assembly
Thyroid	1.5	18.1	1.5×10^{-1}	1.9
Whole Body	5.5×10^{-3}	6.8×10^{-2}	5.7×10^{-4}	7.0×10^{-3}

14.7.4.4 Conclusions

The exclusion boundary doses resulting from a fuel handling accident are within the guidelines of 10CFR Part 100. Thus, a dropped fuel assembly will not present any undue hazard to the health and safety of the public.

14.7.5 SPENT FUEL CASK DROP ACCIDENTS

As discussed in Section 5.4.3.1.9, dropping a spent fuel cask could result in tha rupture of up to 587 intact assemblies. Per Technical Specifications, these assemblies must be decayed for a minimum of 120 days. (Note: A larger number of consolidated fuel rods could rupture, but since these assemblies must be decayed at least 5 years, the dose consequences would be less.) A dose calculation was performed for the assumed rupture of 587 assemblies with 120-day decay. This calculation was performed by ratioing MP2 specific parameters to those generic values used in the dose assessment section of NUREG-0612. The MP2 specific assumptions, which were different from the NUREG-0612 assumptions, were: Power Level - 2700 MW_T, 0-2 HR χ/Q at the EAB - 5.4×10^{-4} sec/m^3, and Number of Assemblies in Core - 217. The resulting whole body dose at the EAB was calculated to be 241 mrem. The thyroid dose is insignificant after 120 days decay. Therefore, the resulting dose is within the acceptable small fraction of 10CFR Part 100 limits.