Docket No. 50-336
B11549

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Proposed Revisions to Technical Specifications

July, 1985

LIMITING CONDITION FOR OPERATION

3.9.17 Prior to movement of a shielded cask over the cask laydown pit, the boron concentration of the pool shall be maintained uniform and sufficient to maintain a boron concentration of greater than or equal to 800 parts per million (ppm).

APPLICABILITY:

Whenever a shielded cask is to be moved over the cask laydown pit.

ACTION:

With the boron concentration less than 800 ppm, suspend all movement of the shielded cask over the cask laydown pit.

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 shielded cask over the cask laydown pit.

REFUELING OPERATIONS

MOVEMENT OF FUEL OVER REGION II RACKS

LIMITING CONDITION FOR OPERATION

3.9.18 Prior to movement of a fuel assembly over a Region II rack 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 is moved over the Region II racks in the spent fuel pool.

ACTION:

With the boron concentration less than 800 ppm, suspend the movement of all fuel over Region II racks.

SURVEILLANCE REQUIREMENTS

4.9.18 Verify that the boron concentration is greater than or equal to 800 ppm within 24 hours prior to any movement of a fuel assembly over a Region II rack in the spent fuel pool, and every 72 hours thereafter.

REFUELING OPERATIONS
SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.19 The Reactivity Condition of the spent fuel pool shall be such that K_{eff} is less than or equal to .95 at all times.

APPLICABILITY:

Whenever fuel is in the spent fuel pool.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.19 Ensure that all fuel assemblies to be placed in Region II (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.

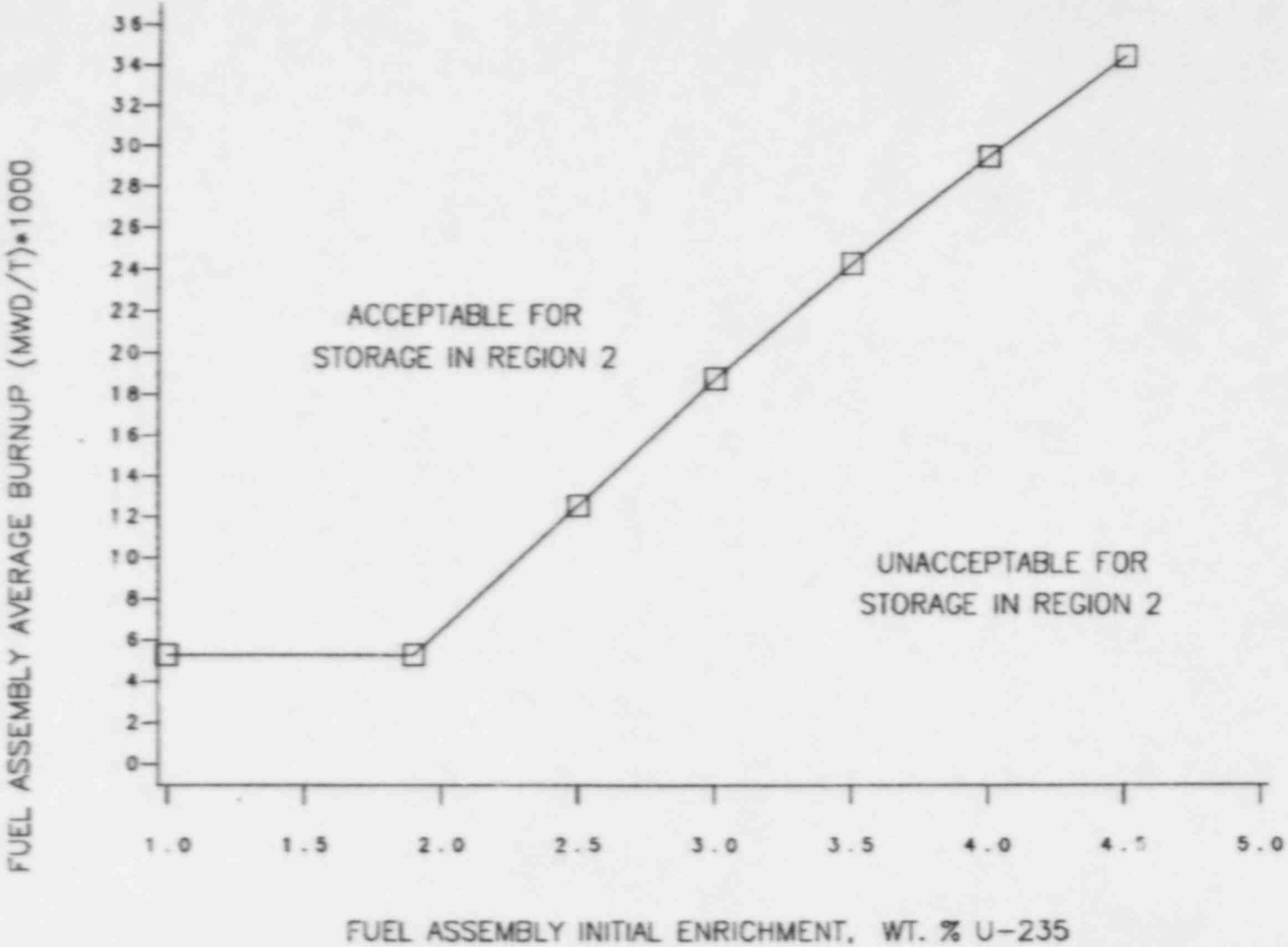
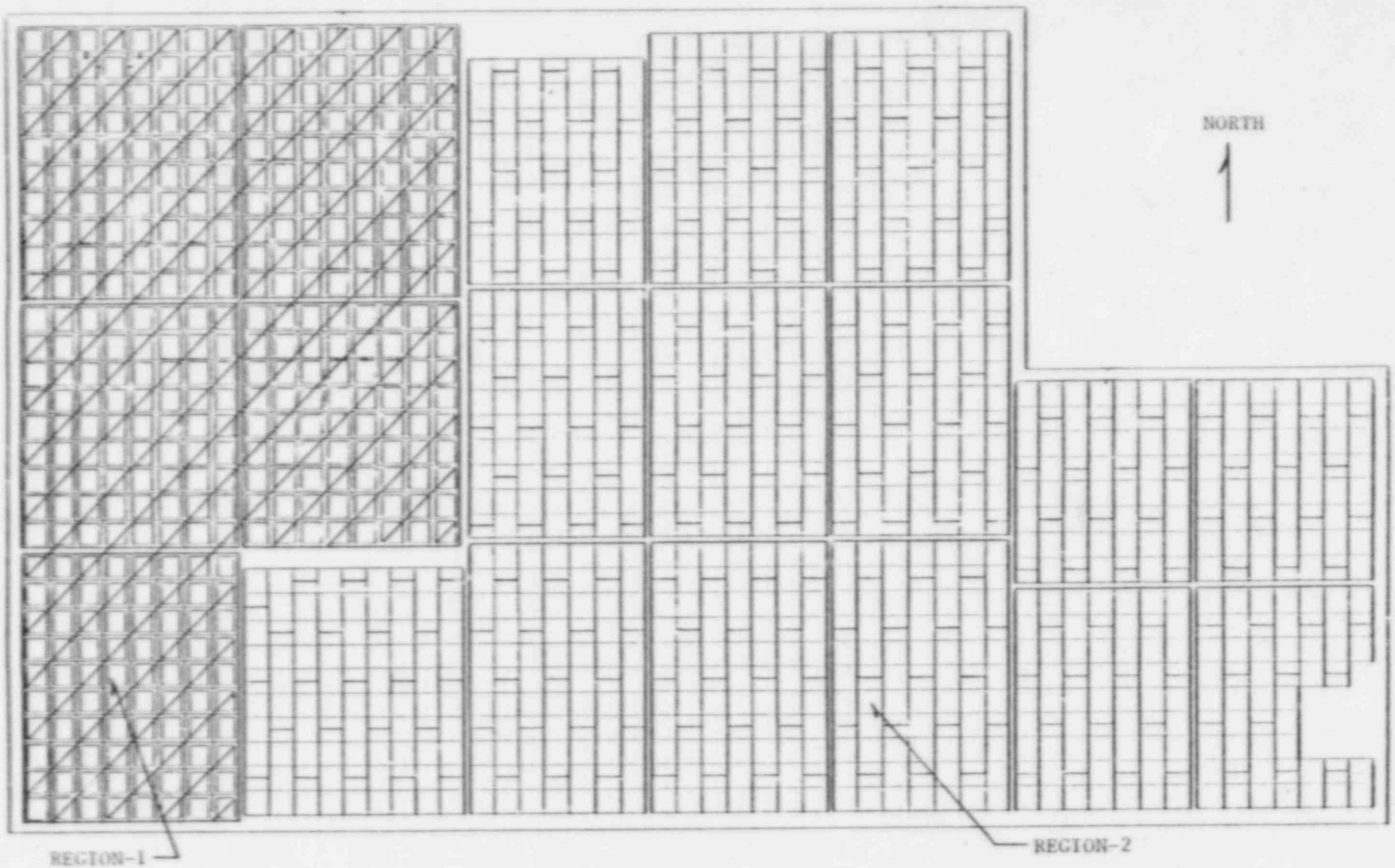


FIGURE 3.9-1 MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2



SPENT FUEL POOL ARRANGEMENT UNIT #2

Figure 3.9-2

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 absorber 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 the doses from ruptured fuel assemblies will be within the assumptions of the safety analyses.

3/4.9.17 SHIELDED CASK

The limitations of this specification ensure that in the event of a cask tilt accident K_{eff} will remain $\leq .95$.

3/4.9.18 MOVEMENT OF FUEL OVER REGION II RACKS

The limitations of this specification ensure that, in the event of a fuel assembly drop accident for a fuel assembly dropped into a Region II rack location completing a 4-out-of-4 fuel assembly geometry, K_{eff} will remain $\leq .95$.

3/4.9.19 SPENT FUEL POOL

The limitations described by Figure 3.9-1 ensure that the reactivity of fuel assemblies introduced into the Region II spent fuel racks 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 fuel enrichment to be stored in these racks is 3.70 weight percent of U-235.

b) Region I 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 may have a maximum fuel enrichment of 4.5 weight percent of U-235.

c) Region II 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 at least 85% of the design burn-up 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 384 storage locations in Region I and 728 storage locations in Region II for a total of 1112 storage locations.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I Items in Section 5.1.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 5.8 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower location shall be as shown on Figure 5.1-1.

5.9 SHORELINE PROTECTION

5.9.1 The provisions for shoreline protection described in Amendments 34, 35 and 36 to the FSAR shall be completed by June 15, 1976.

Docket No. 50-336
B11549

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

Spent Fuel Rerack Safety Analysis Report

July, 1985

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1.0 INTRODUCTION

1.1 LICENSE AMENDMENT REQUEST

This Safety Analysis Report supports a request for an amendment to the Millstone Unit No. 2 Facility Operating License. The amendment request is designed to authorize use of new spent fuel storage racks. The new racks will increase the capacity of the Millstone Unit No. 2 spent fuel pool from 667 storage locations to 1112 storage locations.

1.2 CURRENT STATUS

Northeast Nuclear Energy Company's (NNECO) Millstone Unit No. 2 facility received its operating license in August, 1975. Originally the Millstone Unit No. 2 spent fuel pool had the capacity to store 301 spent fuel assemblies, or about 1.3 full cores. However, by November 1976, NNECO concluded that a capacity expansion of the spent fuel pool was necessary. Backend fuel cycle services, particularly spent fuel reprocessing, had not materialized as anticipated. A spent fuel pool rerack amendment was issued by the NRC on June 30, 1977. The reracking project was completed prior to the first refueling in the fall of 1977. The modified storage pool provided storage locations for 667 fuel assemblies. This amendment provided the capacity needed for the spent fuel discharges plus full core reserve through 1984.

The current pool design configuration is composed of nine rack modules, each containing 63 fuel assembly storage locations in a 7 x 9 array and one rack module containing 100 fuel assembly storage locations in a 10 x 10 array. The modules store the fuel assemblies with a nominal center-to-center spacing of 12.19 inches.

After the Cycle 6 refueling outage in 1985, given the present capacity of 667 storage locations, Millstone Unit No. 2 pool will lose full core reserve capability.(1) As required by the Nuclear Waste Policy Act, 42 U.S.C. 10101, _et. seq._, NNECO has entered into a contract with the Department of Energy for disposal of high level waste and nuclear fuel. However, the Act and the Commission's regulations do not relieve licensees of the responsibility of interim storage of spent nuclear fuel while a repository for ultimate disposal is selected and developed by the Department of Energy. Therefore NNECO is diligently pursuing plans for increasing onsite storage capacity for Millstone Unit No. 2 by replacing the spent fuel pool storage racks with new high density poisoned and non-poisoned racks. The new racks will have a capacity of 1112 storage locations.

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(1) Full core reserve capability is not required by the Atomic Energy Act, the Nuclear Waste Policy Act (NWPA), or the NRC's regulations. NWPA's provisions encourage the maintenance of full core reserve capability and establish it as an objective to be served in determinations as to eligibility for federal assistance with waste disposal. Specifically, under 42 U.S.C. 10155(b) and (g), the absence initially of full core reserve capability could jeopardize eligibility for federal assistance. However, NWPA does not require provisions for full core reserve capability for a plant to continue to operate. _See also_ 10 CFR Part 53; 50 _Fed. Reg._ 5555 at col. 1 (February 11, 1985).

NNECO's proposed modification of the spent fuel pool follows the approach of proposed NRC Regulatory Guide 1.13, Revision 2. This Regulatory Guide permits credit to be taken for reactivity depletion in spent fuel to provide substantially closer center-to-center spacing of fuel assemblies for increased capacity. Additionally, the Regulatory Guide incorporates the concept of a "region strategy" that can be employed to achieve the maximum utilization of the storage space available in the spent fuel pool.

The "region strategy" proposed by NNECO for the Millstone Unit No. 2 spent fuel pool is a two-region dual-pitch design configuration including regions of both poisoned and non-poisoned spent fuel racks. As summarized in Section 2.0, Region I will utilize poisoned racks to accommodate high enrichment, core off-load assemblies in each location in the array. Region II will utilize non-poisoned racks to accommodate, in a 3 out of 4 checkerboard array, fuel assemblies that have sustained at least 85% of design burnup. Combining both regions, the new spent fuel storage racks will consist of 1112 storage locations - sufficient to extend the Millstone Unit No. 2 fuel storage capacity plus full core reserve capacity from June 1985 to 1993. The new spent fuel storage racks will be provided by Combustion Engineering (CE), Inc.

1.3 SUMMARY OF REPORT

Section 2.0 of this report presents a summary of the proposed spent fuel rack design. Sections 3.0 through 5.0 of this report are consistent with the structure and content of Sections III through V of the NRC position paper, "Review and Acceptance of Spent Fuel Storage and Handling Applications", April 1978, as amended by NRC letter dated January 18, 1979.

This report includes discussions of the nuclear, thermal-hydraulic, mechanical, material, structural, and radiological design criteria for the fuel racks. Section 3.0 addresses the nuclear and thermal-hydraulic aspects of the racks. This section addresses the neutron multiplication factor, and considers normal storage and handling of spent fuel as well as postulated accidents. This section also addresses the ability of the spent fuel pool cooling system to maintain sufficient cooling. Section 4.0 addressed the mechanical, material, and structural aspects of the racks. This section concerns the capability of the fuel assemblies, storage racks, and spent fuel pool system to withstand effects of natural phenomena and other service loading conditions. Section 5.0 addresses the radiological or environmental aspects of the racks. These aspects include the thermal, radiological, and chemical releases from the facility under both normal and accident conditions. This section also addresses the occupational radiation exposures and generation of radioactive waste, and includes a cost assessment of the reracking project.

1.4 CONCLUSIONS

The following sections provide information to assist the NRC in its review and approval of the request for an amendment to the Millstone Unit No. 2 Operating License. The modification will help NNECO meet the intent of the Nuclear Waste Policy Act that licensees provide for interim onsite storage of spent fuel.

2.0 SUMMARY OF SPENT FUEL RACK DESIGN

The proposed spent fuel racks are designed to store fuel from Millstone Unit No. 2. These racks have a capacity of 1112 storage locations in two regions of the spent fuel pool.

Region I consists of two 8 x 9 modules and three 8 x 10 modules and would store high-enrichment, core off-load assemblies. The region consists of poisoned spent fuel racks with a nominal center-to-center cell spacing of 9.8 inches. Fuel assemblies would be stored in every location. The five modules of Region I total 384 storage locations and are designed to accommodate 1.7 reactor cores of high enrichment nuclear spent fuel.

The spent fuel rack design for Region I is based upon the commonly accepted physics principle of a "neutron flux trap" with the use of neutron absorber materials. The racks are designed to store Millstone 14 x 14 fuel with an initial enrichment of 4.5 w/o U-235. The poison material to be used is Boroflex.

Region II consists of fourteen modules of non-poisoned spent fuel racks with nominal center-to-center cell spacing of 9.0 inches. The modules consist of 962 cells with useable capacity of 728 storage locations.

Region II is reserved for fuel that has sustained at least 85% of its design burn-up. The spent fuel rack design is based on the criticality acceptance criteria specified in Revision 2 of Regulatory Guide 1.13 which allows credit for reactivity depletion in spent fuel. (Previously, the physics criteria for fuel stored in the spent fuel pool were defined by the maximum unirradiated initial enrichment of the fuel.) Fuel assemblies are stored in a three-out-of-four logic pattern. The fourth location of the storage configuration remains empty to provide the flux trap to maintain the required reactivity control. Blocking devices will be used to prevent inadvertent placing of a fuel assembly in the fourth location.

The spent fuel racks in both regions are fabricated from 304 stainless steel which is 0.135 inches thick. Each cell is formed by welding along the intersecting seams. This enables each spent fuel rack module to become a free-standing module that meets the seismic design requirements without mechanical dependence on neighboring modules or fuel pool walls for support. The rack modules are classified ANS Safety Class III and Seismic Category I.

Both regions of the spent fuel pool have been designed to store fuel assemblies in a safe, coolable, subcritical configuration with K_{eff} less than or equal to 0.95 under the conditions stated in Reference 4-1.

3.0 NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

3.1 NEUTRON MULTIPLICATION FACTOR

Criticality of fuel assemblies in the spent fuel storage rack is prevented by designing the rack to limit fuel assembly interaction. This is done by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies.

The design basis for preventing criticality is, including uncertainties, a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than or equal to 0.95. This criterion is recommended in ANSI N210-1976 and in "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (April 1978), as modified (January 1979).

The following subsections describe: a) the normal and postulated accident conditions in the spent fuel pool which are assumed in calculating the effective neutron multiplication factor (k_{eff}), b) the analysis methodology, and c) the analysis results demonstrating that the design meets the acceptance criterion for criticality.

3.1.1 Normal Storage

a. The analysis considers the most limiting storage condition. In Region I the racks are designed to store fuel assemblies containing 4.5 wt % U-235 in every storage location (Figure 3-1).

   In Region II the fuel is stored in 3 out of 4 locations with cell blocking devices in 1 out of 4 locations (Figure 3-2). In the criticality analysis for Region II, credit was taken for reactivity depletion in the spent fuel consistent with Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis", proposed Revision 2.

b. The moderator is assumed to be pure water at a temperature applicable to the design basis condition which yields the highest reactivity.

c. The Region I and Region II arrays were assumed to be infinite in lateral extent and in length.

d. Mechanical uncertainties (manufacturing tolerances, uncertainty of assembly position in storage racks, material tolerances, etc.) are treated by performing sensitivity studies for the various uncertainties and applying an uncertainty in the k_{eff} value.

e. No control element assemblies (CEAs) or non-contained burnable poisons are assumed to be present.

3.1.2 Postulated Accidents

The double contingency principle of ANSI N16.1-1975 states that it is not necessary to assume concurrently two unlikely independent events to ensure protection against a criticality accident. This contingency principle was applied when considering the following postulated accidents:

1) dropping a fresh fuel assembly into a blocked location of a Region II rack, completing the 4 out of 4 geometry

2) misplacement of a fresh fuel assembly into a Region II rack in one of the 3 open locations

3) dropping a spent fuel assembly into the blocked location of a Region II rack, completing the 4 out of 4 geometry

4) dropping a heavy load (100 Ton cask) into Region II

5) dropping a heavy load (2 Ton gate) into Region I

The bounding accident was determined to be the dropping of a fresh fuel assembly into a blocked fourth location of a Region II rack. To ensure that k_{eff} remains less than or equal to 0.95 for these accidents, the boron concentration in the spent fuel pool will be greater than or equal to 800 ppm during operational conditions under which any of these accidents could occur.

3.1.3 Calculation Methods

The total uncertainty value, as shown below, to be applied to the value of k_{eff} of the storage racks is obtained from the results of the computer codes described in this section.

	delta k_{eff}	
Change	Region I	Region II
Minimum center-to-center distance of storage cells	+0.0130	+0.0038
Maximum Poison Box I.D.	+0.0052	--
Minimum steel thickness	--	+0.0018
Maximum steel thickness	+0.0034	--
Off center placement of fuel assemblies in adjacent cells	+0.0001	-0.0074
Temperature Change	+0.0042	+0.0014
Poison Box not centered in monolith	+0.0014	--
R.M.S. Value	0.0151	0.0086

The individual contributions were combined in a root mean squared manner to yield the values shown as the last entry in the table. When the latter values were then combined in a direct additive manner with the bias and calculational uncertainty, noted above, an overall uncertainty of 0.021 and 0.014 was obtained for Regions I and II, respectively. All components in the overall uncertainty are of at least a 95/95 confidence level.

CEPAK

The CEPAK lattice program is employed to calculate the basic broad group cross section data for the fuel assembly, spent fuel rack structure, and water. This program is a synthesis of a number of computer codes, many of which were developed at other laboratories, e.g., FORM, THERMOS, and CINDER (References 3-1, 3-2, and 3-3). These codes are interlinked in a consistent way with inputs from an extensive library of differential cross section data.

NUTEST

NUTEST is a two-dimensional integral transport code which employs the collision probability technique to compute sub-region dependent reaction rates in an explicit geometric representation of the fuel rods and associated structure of a fuel assembly. This code is used to calculate the flux advantage factors which are applied as correction factors to the basic broad group cross sections computed by the CEPAK lattice program to account for heterogeneous lattice effects not represented in either the multigroup spectrum or homogenized cell spatial calculation, e.g., heterogeneous fast fission effect in fuel pellets.

DOT-2W

The spatial flux solution and multiplication factor for an infinite array of individual or clusters of fuel storage cells are computed with the two dimensional, discrete ordinates transport code, DOT-2W (Reference 3-4). The major features of the method used in this code are:

a) Energy dependence is considered using the multigroup treatment.

b) The derivative terms and spatial dependence are approximated using a finite difference technique.

c) Dependence upon the direction variables is treated using the discrete ordinates method.

d) The scattering integral is evaluated using a discrete ordinates quadrature in combination with a Legendre expansion of the scattering kernel to approximate anisotropic scattering.

3.1.4 Acceptance Criterion for Criticality

The acceptance criterion for the neutron multiplication factor (k_{eff}) is that it be less than or equal to 0.95, including uncertainties, under all postulated conditions. For Region I the resulting neutron multiplication factor (k_{eff}) is 0.939 including all uncertainties and calculational biases. For Region II reactivity depletion is a function of the percentage of burnup achieved, not of the initial enrichment. The resulting reactivity, including uncertainties and calculational biases, as a function of burnup for several initial enrichments is shown in Figure 3-3. The minimum allowable burnup for a given initial enrichment is that corresponding to k_{eff} = 0.95. When these minimum burnup values are plotted as a function of initial enrichment, regions of acceptable and unacceptable burnup are identified (see Figure 3-4).

3.1.5 Thermal-Hydraulic Analyses

The thermal-hydraulic analyses for the fuel storage racks are based on the design parameters given in Table 3-1. It is assumed in the analyses that the fuel pool is loaded to full capacity with intact fuel assemblies. The spent fuel racks are designed to adequately cool the spent fuel during normal and accident conditions subject to the following criteria:

1. Bulk boiling of the entire pool must not exist during normal operation.

2. Maximum fuel clad temperature will not exceed 650°F during both normal operation and accident conditions.

Normal operation includes any arrangement of intact fuel assemblies with bulk pool temperature of 150°F and a minimum pool depth of 23 feet of water above the fuel. Two accident conditions were considered. In the first, it is assumed that, as a result of loss of external cooling, coolant is evaporated to a minimum pool depth of 10 feet of water above the racks. For the second case, it is assumed that, as a result of cavity seal failure and resulting spent fuel pool drain down, coolant is evaporated to a minimum pool depth of two feet above the top of the active fuel. The results of the thermal-hydraulic analysis confirm that the above two design criteria are met.

TABLE 3-1

THERMAL HYDRAULIC DESIGN PARAMETERS

Number Assemblies Normal Reload	1/3 core
Minimum Decay Time One Assembly	3 days
Minimum Decay Time Full Core	6 days
Water Height Above Fuel	23 feet
Minimum Water Height Above Rack (Accident)	10 feet
Maximum Bulk Water Temperature (Normal)	150°F
Maximum Water Temperature in the Rack Region (Accident)	235°F

3.2 COOLING CONSIDERATIONS

3.2.1 General Description

The fuel pool cooling and cleanup system consists of two circulating pumps with a capacity of 850 gpm each, and two heat exchangers with a total heat removal capacity of 11.3×10^6 BTU/hr. The system was originally designed to maintain the fuel pool temperature below 120°F during the normal heat load condition. With the aid of the shutdown cooling system, the design temperature limit of 150°F can be maintained under the maximum heat load conditions. The maximum heat load was defined in the FSAR as 28.0×10^6 BTU/hr. The maximum heat load conditions for the enlarged fuel pool capacity are defined in Section 3.2.2. The disposal cartridge type fuel pool filters and a demineralizer maintain water quality. This system is described in Section 9.5 of the FSAR.

3.2.2 Cooling System Performance

The adequacy of the cooling system has been analyzed in view of the expanded fuel storage capacity. Table 3-2 summarizes the cooling system performance for the normal refueling and full core offload conditions.

TABLE 3-2

SPENT FUEL POOL COOLING SYSTEM
HEAT LOADS AND OPERATING TEMPERATURES

Normal Maximum Heat Load: 15.2×10^6 BTU/hr(1)

Abnormal Maximum Heat Load: 37.8×10^6 BTU/hr(2)

OPERATING CONDITION	POOL TEMPERATURES	
Design Basis	Design	Calculated
Normal Maximum	120°F	131°F(3)
Abnormal Maximum	150°F	120°F(4)
Single Active Failure of a SFP Cooling Train		
Normal Maximum	212°F	178°F
Abnormal Maximum	212°F	185°F
Total Loss of Forced Pool Cooling		
Normal Maximum	212°F	9 3/4 hrs. to boiling
Abnormal Maximum	212°F	4 hrs. to boiling

(1) This heat load is predicted for normal refueling with the most recently unloaded one-third core having decayed for 150 hours. After 12 more days, the decay heat load will be less than 11.3×10^6 BTU/hr.

(2) This heat load is predicted for spent fuel in the pool with the entire core offloaded.

(3) This temperature is a function of using only the Spent Fuel Pool cooling heat exchangers.

(4) This temperature value is a function of using one train of the Shutdown Cooling Heat Exchangers in addition to Spent Fuel Pool Heat Exchangers.

The decay heat loads were calculated using the computer code ORIGEN, developed at Oak Ridge National Laboratory. ORIGEN is a point depletion code which solves the equations of radioactive buildup and decay for large numbers of isotopes.

The design basis heat load was determined by the following refueling sequence:

1. 868 fuel assemblies present in the pool after the 1992 refueling.

2. 217 fuel assemblies (full core) offloaded during the refueling shutdown in 1993.

The projected capacity of the fuel pool is 1112 assemblies, however, the twenty-seven "spare" assembly spaces were assumed to contain assemblies from the first refueling.

ORIGEN predicts 37.8×10^6 BTU/hr. for the design basis heat load. The combined cooling capability of the Spent Fuel Pool Heat Exchangers and one train of the Shutdown Cooling Heat Exchangers is 38.5×10^6 BTU/hr. The size of the original fuel pool cooling system as described in the FSAR was based on infinite irradiation time for the fuel assemblies, whereas the ORIGEN heat load was based on finite irradiation time and the actual refueling schedule.

A single failure analysis of the spent fuel pool cooling system is presented in Table 3-2.

3.2.3 Fuel Element Heat Transfer

An analysis of the maximum fuel cladding temperature was performed for the postulated case of complete loss of coolant circulation to the pool. The analysis assumed maximum anticipated heat load in the pool, with the hottest assembly located in the least cooled storage area. The maximum cladding temperature will occur at the location of maximum heat flux. Natural circulation flow rates within storage tubes will result in a heat transfer coefficient in excess of 50 BTU/hr. ft.^{2} ^{o}F. Because the heat flux is small, very large uncertainties in the film coefficient are acceptable without causing prohibitively high clad temperatures. The design upper limit temperature for the clad in the spent fuel pool is 650^{o}F. The fuel cladding temperature analysis verifies that this limit is maintained for the Millstone Unit No. 2 racks.

3.2.4 Spent Fuel Pool Chemistry Control

Water chemistry and optical clarity will be maintained by the existing spent fuel pool cleanup system. The cleanup system is a non-safety related system and has been designed to non-Seismic Category I requirements. Isolation capabilities from the Category I portion of the fuel pool cooling system have been provided by Seismic Category I

isolation valves. The cleanup system consists of two refueling water purification pumps, two filters, and a demineralizer and associated valves and piping. In addition, the spent fuel pool has been provided with a skimmer pump and two filter assemblies to facilitate the removal of accumulated surface debris. The cleanup system has been designed to process water through the purification loop from the refueling pool and refueling water storage tank.

The radioactive contaminant levels in the pool are primarily a function of failed fuel fraction and reactor operating level and are highest during and shortly following refuelings. Since the concentration of impurities is controlled by continuous removal in the demineralizer and by natural radioactive decay, the increase in contaminant levels due to the long-term stored fuel is negligible when compared to the removal mechanisms. Therefore, the installed systems and equipment are deemed adequate for maintaining water chemistry and optical clarity with the expanded pool storage capacity.

3.3 POTENTIAL FUEL AND LOAD HANDLING ACCIDENTS

3.3.1 Fuel Handling Accident in the Spent Fuel Pool

This accident has been addressed in the Millstone Unit No. 2 FSAR (Section 14.19). A complete list of assumptions is provided in FSAR Table 14.19-1. Results of the analysis, which are well below the limits of 10 CFR Part 100, are presented in Section 14.19.3.

3.3.2 Load Handling

The reracking operation at Millstone Unit No. 2 will be conducted in accordance with strict procedures to prevent inadvertent dropping of heavy objects into the pool during the reracking operations. Strict procedural controls as well as Technical Specifications will prohibit the movement of heavy objects over spent fuel stored in the pool.

During the reracking, no heavy loads supporting the activities in the spent fuel pool will have a handling path that brings them directly above or in the immediate vicinity of the stored spent fuel. Additionally, during the reracking transition, no fuel racks old or new will have a handling path directly over stored spent fuel.

Safe load paths have been established which will prevent heavy loads from being transported over spent fuel. Additionally, all lift rigging, hoists, and cranes will be designed and operated in accordance with C.M.A.A. Specification No. 70, ANSI B30.2, B30.9, and B30.11 respectively. All heavy lifts will be performed in accordance with established station procedures, which have been revised to comply with NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants." This will minimize the possibility of a heavy load drop accident.

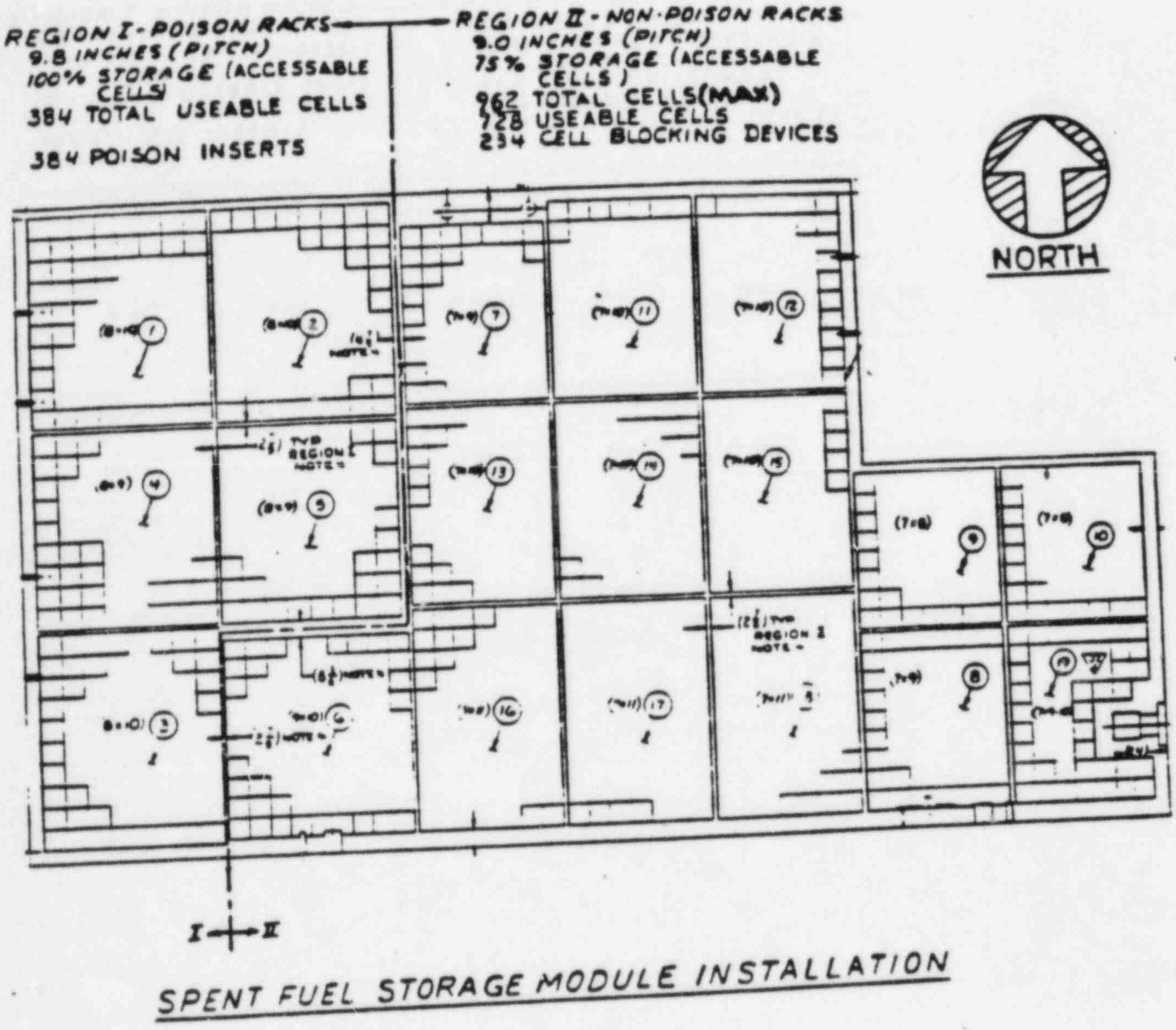

FIGURE 3-1

Fuel Pool Arrangement: Two Region

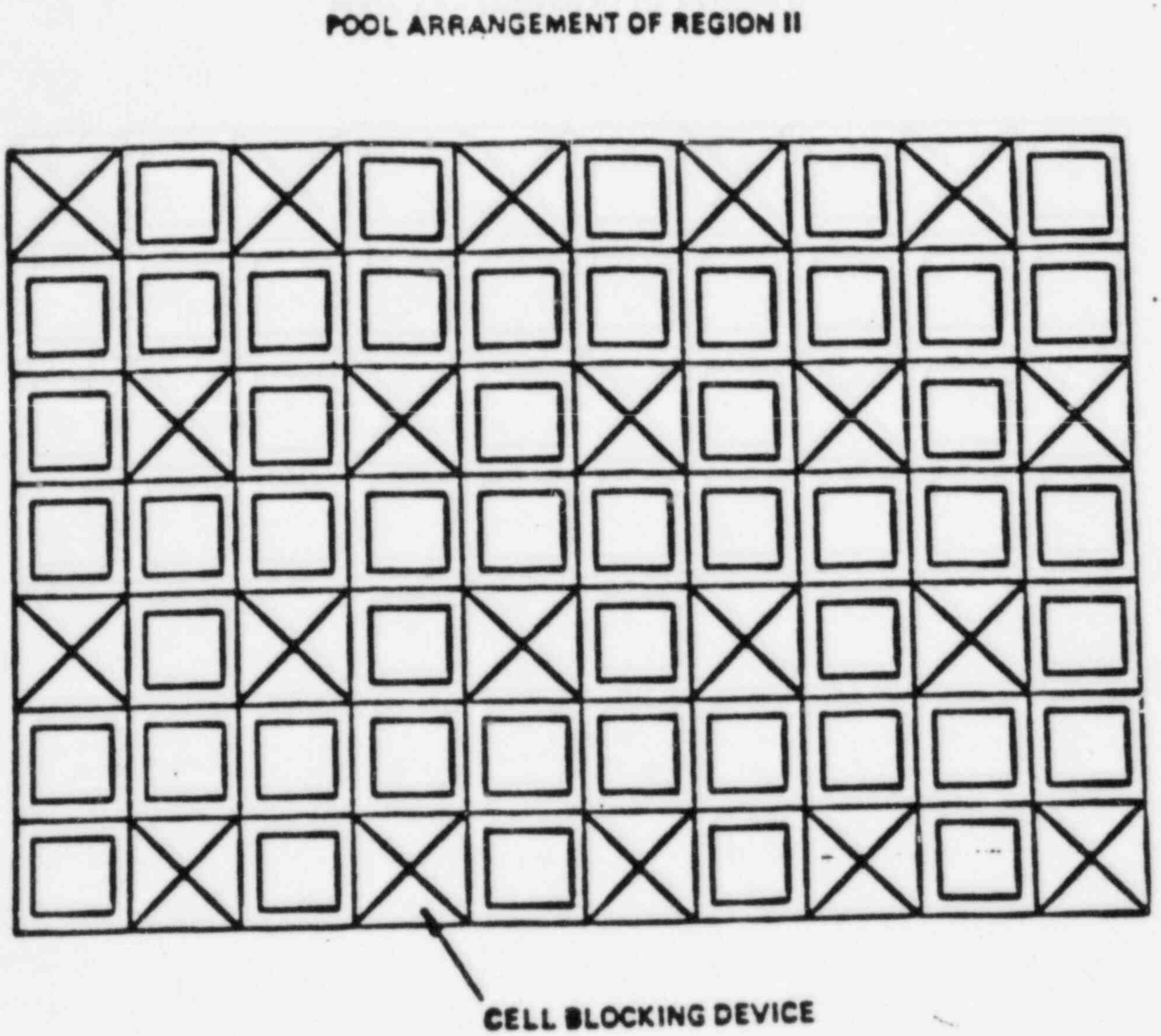

FIGURE 3-2

Spent Fuel Rack Storage Configuration
For Region II

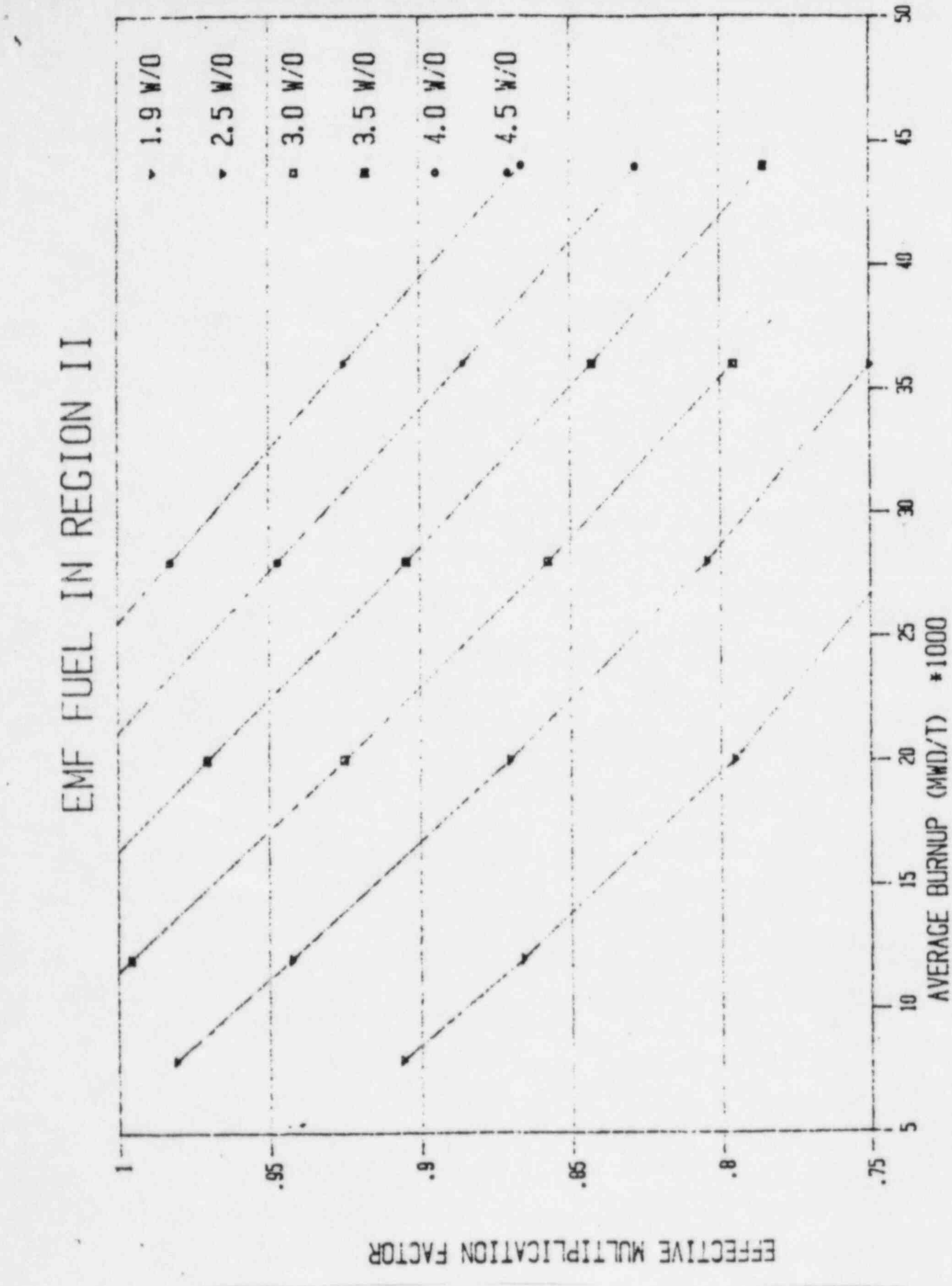

FIGURE 3-3

Effective Multiplication Factor As
Function of Burnup of Region II

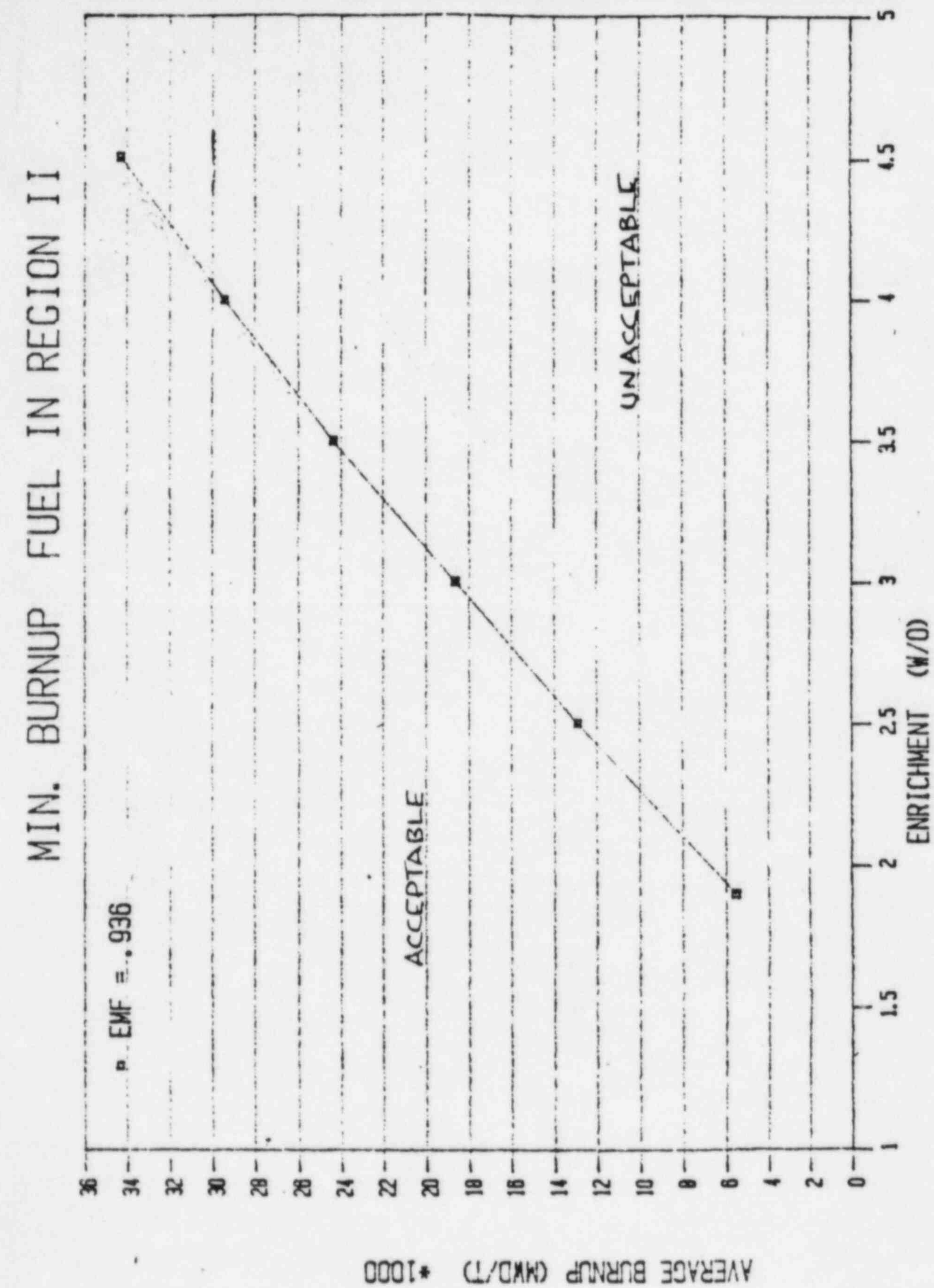

FIGURE 3-4

Minimum Allowable Burnup as Function of
Initial Fuel Enrichment for Region II

3.4 REFERENCES

3-1 FORM - A Fourier Transform Fast Spectrum Code for the IBM-7090, McGoff, D. J., NAA-SR-Memor 5766, September, 1960.

3-2 THERMOS - A Thermalization Transport Theory Code for Reactor Lattice Calculations, Honeck, H., BNL-5816, July, 1961.

3-3 CINDER - A One Point Depletion and Fission Product Program, England, T. R., WAPD-TM-334, Revised June, 1964.

3-4 R. G. Soltesz, et. al., "Users Manual for DOT-2W Discrete Ordinates Transport Computer Code," WANL-TME-1982, December, 1969.

4.0 MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

4.1 DESCRIPTION OF STRUCTURE

4.1.1 Description of the Auxiliary Building

The auxiliary building is a multi-story, reinforced concrete structure with flat slabs and shear walls. Some open areas of the building are supported by structural steel columns to preserve space and allow flexibility in the design. The portion of the building west of column line M.7 is founded on bedrock approximately 60 feet below the ground surface, while the eastern end of the building is supported by compacted structural backfill. These two portions of the building are separated from each other by an expansion joint at line M.7, to allow for differential movements.

Spent fuel storage is provided between column lines 17.2 and 18.9 and column lines H.4 and L.5 at Elevation (-)2'-0". The storage area consists of a reinforced concrete pool lined with one-fourth inch thick stainless steel plate to Elevation 38'-6". Normal water level is to Elevation 36'-6".

4.1.2 Description of Spent Fuel Racks

4.1.2.1 Design and Fabrication of Spent Fuel Racks

The spent fuel storage racks are fabricated with 304 stainless steel having a maximum carbon content of 0.065%. The racks are monolithic honeycomb structures with square fuel storage locations as shown in Figure 4-1. Each storage location is formed by welding stainless steel sections along the intersecting seams, permitting the assembled cavities to become the load bearing structure, as well as framing the storage cell enclosures. Each module is free standing, and seismically qualified without mechanical dependence on neighboring modules or pool walls. This feature allows remote installation (or removal if required for pool maintenance). Reinforcing plates at the upper corners provide the required strength for handling.

Stainless steel bars, which are inserted horizontally through the rectangular slots in the lower region of the module, support the fuel assemblies. The support bars are welded in place and support an entire row of fuel assemblies. The module is supported by adjustable pads to facilitate leveling at installation.

Loading of the fuel racks is facilitated via movable lead-in funnels. The openings of the funnels are symmetrical. The funnels are placed on top of the rack module.

The 304 stainless steel module wall thickness is 0.135 inch.

Region I is located within 5 modules and comprises a total of 384 cavities. Region I is the high-enrichment, core off-load region. The fuel assemblies can be stored in every location.

Region I is designed for a total of 384 usable cavities (i.e., 100% storage) for enrichment up to and including 4.5 w/o U-235. Each cavity in Region I contains a poison insert. These inserts are made up of Boroflex sheets enclosed, but not sealed, between two stainless steel sheets. The neutron absorbing material, Boraflex, is designed to be compatible with the pool environment. Positive venting of the poison is provided by the 3/8" diameter hole in the inner walls of the inserts. (Figure 4-5). The stainless steel sheets are configured such that a minimum water gap is maintained between the insert and the cell wall. These inserts lock into the storage cavity using a spring locking mechanism on the upper end which snaps into a hole in the surrounding cell wall. These poison inserts are neutron absorbers. A typical Region I fuel rack module and poison insert are shown in Figures 4-1 and 4-4.

The neutron material, Boraflex, used in the Region I spent fuel racks is manufactured by Brand Industrial Services, Inc., and fabricated in accordance with the quality assurance criteria of 10 CFR 50, Appendix B. Boraflex is a silicone based polymer containing fine particles of boron carbide in a homogeneous, stable matrix. Boraflex contains a minimum ^{10}B areal density of 0.030 gm/cm^2 for Region I racks.

Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments and to verify its structural integrity and suitability as a neutron absorbing material. Tests were performed at the University of Michigan exposing Boraflex to 1.03×10^{11} rads gamma radiation with a substantial concurrent neutron flux in borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities before and after being subjected to an environment of borated water and 1.03×10^{11} rads gamma radiation.

Long term borated water soak tests at high temperatures were also conducted. It was shown that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or softening. Boraflex maintains its functional performance characteristics and shows no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

The actual tests verify that Boraflex maintains long-term material stability and mechanical integrity and can be safely utilized as a poison material for neutron absorption in spent fuel storage racks.

The inservice poison material surveillance program is addressed in Section 4.7 of this submittal.

Region II consists of a total of 962 cavities. Within Region II, fuel can be stored in 75% of the total available cavities (see Figure 4-2) for a

storage capacity of up to 728 cavities. Cell blocking devices are used to preclude placement of fuel into every fourth cavity, which remains empty and provides a flux trap for reactivity control. Figure 3-1 shows the arrangement of Region I and Region II modules in the Millstone Unit No. 2 pool.

4.1.2.2 Support of Spent Fuel Racks

The spent fuel racks have been designed for direct bearing onto the spent fuel pool floor. An adjustable pad (Figure 4-3) is provided under each corner of the fuel rack. Fuel rack module leveling is accomplished by adjusting each pad height to conform to the pool floor. Each foot can be raised or lowered 1/4" and can rotate 2 to 3 degrees to accommodate pool floor variations during installation.

4.1.2.3 Fuel Handling

The design of the spent fuel racks will not affect the conclusions of the fuel handling accidents presented in the FSAR and summarized by the NRC in the Safety Evaluation Report. That is, the radiological doses for the postulated fuel assembly drop accident are well within the criteria of 10 CFR Part 100.

4.2 APPLICABLE CODES, STANDARDS, AND SPECIFICATIONS

The spent fuel racks are designed in accordance with the following codes, standards, and specifications:

1. Code of Federal Regulations, 10 CFR Part 50:

   a) Appendix A, "General Design Criteria for Nuclear Plants," Criteria 2, 3, 4, 5, 61, 62, 63.

   b) Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

2. ASME Boiler and Pressure Vessel Code Section III, Subsection NF, "Nuclear Power Plant Components."

3. ASME Boiler and Pressure Vessel Code Section IX.

4. American Society for Testing Materials Documents:

   a) ASTM - A240 - Specification for Corrosion Resisting Chromium Nickel Steel Plate, Sheet & Strip for Fusion-Welded Unfired Pressure Vessels.

   b) ASTM - A276 - Specification for Stainless and Heat Resisting Bars and Shapes.

5. American National Standards Institute:

    a) ANSI - N210, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations, 1976.

    b) ANSI - N16.1, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, 1975.

6. United States Nuclear Regulatory Commission:

    a) Standard Review Plan, Section 9.1.2, Rev. 2 "Spent Fuel Storage."

    b) Regulatory Guide 1.13, Rev. 2 Draft, "Spent Fuel Storage Facility Design Basis."

    c) Regulatory Guide 1.26, Rev. 3 "Quality Group Classification and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants."

    d) Regulatory Guide 1.29, Rev. 3 "Seismic Design Classification."

    e) Regulatory Guide 1.31, Rev. 2 "Control of Stainless Steel Welding," as modified by Branch Technical Position MTEB-51, "Interim Position on Regulatory Guide 1.31, 'Control of Stainless Steel Welding'."

    f) Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," Rev. 1, February 1978.

    g) Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Rev. 3.

    h) NRC Guidance, "Review and Acceptance of Spent Fuel Storage and Handling Applications" April 1978, and Modifications dated January 18, 1979.

4.3 SEISMIC AND IMPACT LOADS

Maximum loads transmitted to the floor by the spent fuel racks at Millstone Unit No. 2 are given below.

Floor-Rack Interface Loads (lb_F/Pad)

	North-South	East-West	Vertical (Down)
OBE + Dead Weight (4 pads in contact)	12,100	12,100	107,300
SSE + Dead Weight (2 pads in contact)	39,100	38,900	131,200
SSE + Dead Weight + Vertical Impact (4 pads in contact)	39,100	38,900	133,000

The seismic analysis of the spent fuel rack includes an assessment of the maximum sliding and tipping that can be expected. The racks are installed with a nominal gap of 2" between modules and are a minimum of 4-13/16 inches from the pool walls. The analysis has shown that the maximum motion of the racks, including tipping, sliding, and thermal expansion is less than the gap between adjacent modules. Therefore, no contact is predicted.

4.4 LOADS, LOAD COMBINATIONS, AND STRUCTURAL ACCEPTANCE CRITERIA

4.4.1 Spent Fuel Pool Rack Analysis

The loads, load combinations, and structural acceptance criteria used in the structural analysis of the spent fuel racks are listed below and are consistent with the NRC guidance in "Review and Acceptance of Spent Fuel Storage and Handling Applications" (Reference 4-1).

Load Combination (Elastic Analysis)	Acceptance Limit
D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + To	Lesser of 2Sy or Su stress range
D + L + To + E	Lesser of 2Sy or Su stress range
D + L + Ta + E	Lesser of 2Sy or Su stress range
D + L + Ta + E'	Faulted Condition Limits of NF 3231.1c

The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined.

4.4.2 Pool/Auxiliary Building Analysis

The Millstone Unit No. 2 spent fuel pool rerack program requires an analysis of the spent fuel pool and auxiliary building for the increased loads caused by the increased storage of spent fuel. This section addresses the guidelines and acceptance criteria that NNECO followed to qualify the spent fuel pool and auxiliary building for the new loadings.

The Millstone Unit No. 2 auxiliary building is a multi-story concrete structure. Spent fuel storage is provided between column lines 17.2 and 18.9 and column lines H.4 and L.5 at elevation (-)2' -0". The storage area consists of a reinforced concrete pool lined with a one-quarter inch thick stainless steel liner to elevation 38'-6". Normal water level is to elevation 36'-6". A leak chase system consisting of channels embedded behind the liner at all seams and connected to a collector system is used to monitor and control any possible leak from the pool. Construction materials used in the construction of the pool include ASTM A-240, Type 304 stainless steel, ASTM A-615 Grade 60 reinforcing steel, and 3,000 psi 28-day strength concrete.

The analysis of the spent fuel pool and associated components of the auxiliary building to accommodate the loadings associated with increased storage capacity was accomplished with the use of a large finite element model. The finite element model included the entire spent fuel pool, fuel transfer canal, and cask laydown area. The foundation of the pool was included and terminated at points where it was considered that the local effects are negligible with regard to the overall pool response. The effect of the auxiliary building floors and walls that frame into the pool was included so that advantage could be taken of the stiffening action and load resisting capacity that this structural system provides.

The spent fuel pool analysis proceeded with the formation of composite load cases. The composite loads consisted of a combination of basic load cases which were grouped together for the purpose of application of the

Standard Review Plan load factors. The composite load cases included dead load, live load, operating thermal and accident thermal, and Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) loads. Dead load consisted of a combination of the dead weight of concrete, hydrostatic pressure, and the weight of the fuel rack modules without their fuel components. Live loads consisted entirely of the submerged weight of the fuel assembly in its rack location. Normal operating thermal loads place the pool water at 150°F with the temperature outside the pool at 55°F. The 212°F pool/wall interface temperature for accident thermal was deemed applicable when determining the gross structural effects on the pool walls. OBE earthquake loads involved four composite load cases. These load cases result from the specification in the Standard Review Plan that the three directions of earthquakes must be applied simultaneously and permutation of signs must be included. For this reason four load cases were specified with appropriate permutations of signs of the three orthogonal accelerations. An additional four load cases were developed by multiplying each of the originals by (-)1.0. Load combinations involving SSE earthquake loads utilized these OBE earthquake loads with a coefficient applied.

The Standard Review Plan specifies service load and factored load combinations for Category 1 concrete structures. Upon examination of these load cases, it can be shown that eight of the composite loads comprising them are not under consideration when analyzing the spent fuel pool. Other accident loads such as flood loads, tornado loads, or any piping loads were included, consistent with their definition and in the manner in which they affect the spent fuel pool structure.

Upon examination of the Standard Review Plan load combinations, it is readily apparent that some of our load cases are duplicates or envelop other load cases. These combinations were reviewed and controlling load combinations were chosen from this group. Following the load combinations, the concrete sections were checked against criteria set forth in the latest revision of the American Concrete Institute Code Requirements for Nuclear Safety Related Concrete Structures - ACI 349-80.

Cask drop loads have been addressed in the Millstone Unit No. 2 FSAR. Guidelines set forth for the areas of the pool where the cask may be safely handled remained as stated.

In sum, by following the above guidelines and acceptance criteria, NNECO demonstrated the adequacy of the Millstone Unit No. 2 spent fuel pool and auxiliary building to accommodate the loads associated with the increased spent fuel storage capacity.

4.5 DESIGN AND ANALYSIS PROCEDURES

4.5.1 Methodology Summary

The spent fuel storage racks are designed to withstand forces generated during normal operation, an Operating Basis Earthquake, or a Safe Shutdown Earthquake. Lateral and vertical seismic loads along with fluid forces are considered to be acting simultaneously on the fuel racks. The racks are designed to assure rack structural integrity while at the same time keeping the fuel in a subcritical state.

Linear response spectrum methods are used for the vertical direction. The lateral seismic responses of the spent fuel storage racks are determined using a non-linear time history analysis. Non-linear time history analyses are performed for the lateral directions primarily because of fuel impacting. The effects of impacting structures significantly influence the stresses in both the storage structure and the fuel and, because they are non-linear in nature, can only be accounted for by performing more complex non-linear time history analyses. Rack vertical impact loads on the floor due to tipping are calculated.

The seismic input used for these analyses consists of the vertical response spectrum and the lateral acceleration time histories corresponding to the pool floor elevation at Millstone Unit No. 2. The analyses are performed in accordance with Regulatory Guide 1.122, Revision 1, February 1978.

The first step in the analytical procedure is to determine the dynamic characteristics of the fuel storage racks. This is done by developing a three-dimensional finite element model of the structure and solving for the natural frequencies and mode shapes in air. The finite element code used in the study is SAP IV (see Section 4.5.2).

The resulting dynamic characteristics are then incorporated into a non-linear representation of the entire system, which includes the fuel and the storage racks. The CESHOCK computer code (see Section 4.5.2), is used to determine the non-linear time history response of the system. The effects of impacting between the fuel and the storage rack are represented in the CESHOCK model. Because of the close proximity of the structures, hydrodynamic coupling effects between the fuel, the storage rack, and the pool are also included in the model. (See Reference 4-2 for additional information.)

The racks are analyzed using a finite element model in the SAP IV code and the loads from Section 4.3. SAP IV output consists of membrane stresses and bending moments for each element.

The component stress on each element resulting from the application of each directional load is combined by the square root sum of the squares method. The results are compared to stress allowables in accordance with the rules of ASME Boiler & Pressure Vessel Code, Section III, Subsection NF.

Structural analyses were also performed to evaluate the results of a maximum crane uplift force and fuel assembly drop on the fuel storage rack structure.

A fuel assembly falling onto the racks can either fall into a cavity or onto the top of the racks. In the former case, the impact will not affect the primary function of the racks which is to maintain separation between bundles and insure the flow of coolant. In the latter case, the load resulting from the impact of a fuel bundle dropped from its maximum lifted height was calculated. Using a finite element model the calculated impact force was applied at various locations on the top of the rack.

The resultant stress was determined to be less than the material yield stress at the elevation of the active fuel, therefore, it is also concluded that the racks' primary function remains unaffected.

The maximum load that can be applied by the fuel handling crane prior to initiating an audible alarm is 2000 lbs. (rated capacity of 1 ton). For the analysis a maximum load of 3000 lbs. was assumed. This load was applied to the fuel rack; resulting stresses were well within allowable limits.

4.5.2 Computer Code Descriptions

The computer codes used in these analyses are described in the following subsections.

SAP IV

SAP IV is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve for the displacements and compute the stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plane strain-plane stress, thick shell, spring, axisymmetric elements. The program can treat thermal and various forms of mechanical loading as well as internal element loadings. Dynamic analysis options consist of eigenvalue solutions yielding frequencies and mode shapes, response history by mode superposition, response history by direction integration, and response spectrum analysis. Earthquake type of loading as well as time varying pressure can be treated. The output consists of displacements at each nodal point as well as internal member forces for each element.

CESHOCK

The CESHOCK computer code performs transient, dynamic analyses of non-linear elastic systems. These systems can be either axial models having one degree-of-freedom per node or lateral ones having one rotational and one translational degree of freedom per node. The response of a system is determined by numerically integrating (using a Runge-Kutta-Gill technique) its equations of motion. Excitation can

take the form of either initial conditions or time histories of applied accelerations, velocities displacements or forces. The non-linearities can consist of gaps, friction, hysteresis or non-linear springs. Hydrodynamic action can also be modeled, with both on-diagonal (added mass) and off-diagonal (coupling) terms being considered.

The program automatically searches the response time histories and prints out the maximum and minimum values of all nodal accelerations and member loads, and can generate an optional output tape containing the complete response histories.

CESHOCK is an extensively modified, proprietary version of the SHOCK computer code developed by V. K. Gabrielson and R. T. Reese of Sandia Laboratories (Reference 4-4). If differs from the original in the areas of damping, coefficient of restitution, friction, hydrodynamic effects, hysteresis, input of time histories, output options, allowable problem size, and the manner of inputting stiffness elements. CESHOCK has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted analytical results via an independent computer code (References 4-4 and 4-5).

4.6 MATERIALS AND QUALITY CONTROL

4.6.1 Materials

The following provides a list of materials used in the construction of the spent fuel racks:

FUEL STORAGE MODULE:

A. Basic Module:	ASTM-A240-304 Stainless Steel, max. carbon content, .065%
B. Support Bars:	ASTM-A276 or A-479-304 Stainless Steel, max. carbon content .065%.

ADJUSTABLE FOOT:

A. Block:	ASTM-A182 max. carbon content .065%.
B. Retaining Ring and Pad:	ASTM-A240-304 Stainless Steel, max. carbon content .065%.
C. Screw:	ASTM-A453 grade 660, high alloy steel.

POISON BOX

A.	Inner/Outer Shell, Funnel, Base and Braces:	ASTM-A240-304 Stainless Steel, max. carbon content .065%.
B.	Insert:	AMS 5697, in the quarter hard condition Rockwell "C" -23, max. carbon content .065%.
C.	Poison:	Boraflex - Bisco Products Inc. - Supplier.

CELL BLOCKING DEVICES

A.	Pipe:	ASTM-A312 304 Stainless Steel, max. carbon content .065%.
B.	Pin:	ASTM-A479 304 Stainless Steel, max. carbon content .065%.
C.	Detent Pin:	300 Series Stainless Steel.

4.6.2 Quality Control

Northeast Utilities' and Combustion Engineering's Quality Assurance Programs ensure that all manufacturing and installation activities conform to acceptable quality requirements throughout all areas of performance. The pertinent requirements of 10 CFR Part 50, Appendix B, and Combustion Engineering Quality Assurance Topical Report, CENPD-210-A, Rev. 3, and Specification 00000-WQC-5.2 will be followed. In addition, Northeast Utilities' Topical QA Report, Revision 6, describes Quality Assurance requirements with which the design, procurement, and fabrication of the new fuel storage racks will comply.

4.7 POISON MATERIAL IN-SERVICE SURVEILLANCE PROGRAM

A long term surveillance program will be implemented to ensure continued acceptable performance of the Boraflex neutron poison material used in the racks. This program will be based on current performance information for the Boraflex neutron poison material.

Proper documentation will be obtained from the manufacturers of Boraflex and the racks to assure the quality of the neutron poison material and its proper loading in the racks. Visual inspection of the racks will be performed upon receipt of the racks to verify that the Boraflex is loaded in each of the specified locations in each rack.

Uniquely identified samples, taken from material representative of that used as a neutron absorber, will be placed within surveillance capsules

and placed within the Region I racks. At specified intervals the samples will be removed and tested for boron content.

Irradiation tests have been previously performed to determine the stability of Boraflex in boric acid solution. The results of these tests are documented in test reports of the Bisco Corporation (References 4-6, 4-7, 4-8). From these tests, there is no evidence indicating any deterioration of the Boraflex material through a cumulative irradiation in excess of 1×10^{11} rads gamma affecting the suitability of Boraflex as a neutron poison material. Under the proposed surveillance program, calculations have shown that the specimens would require at least five years in the pool environment to approach this level of cumulative exposure. Direct dosimetry will be utilized, however, to establish an accurate record of cumulative exposure. Periodic testing and examination of poison specimens will take place beyond the point at which cumulative exposures exceed those of documented tests.

The surveillance specimens will initially be examined after approximately five years of exposure in the pool environment. Several specimens will be checked for mechanical integrity and other general performance characteristics. This examination will include visual inspection as well as other tests to verify the material stability. This initial surveillance will be used to verify that the performance of the Boraflex is consistent with current Bisco Corporation test results. Based on the results of the initial surveillance, and results from existing fuel rack surveillance programs at the Millstone Nuclear Station, additional testing will be scheduled to assure acceptable material performance throughout the life of the plant.

4.8 REFERENCES

4-1 NRC Guidance, "Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 1978 and modified January 19, 1979.

4-2 Longo, R., and Baisley, D. F., "Seismic Analysis of Spent Fuel Racks," ANS paper TS-7308, presented at the ANS Topical Meeting on Options for Spent Fuel Storage at Savannah, Georgia, September 26-29, 1982.

4-3 Bathe, K. J., Wilson, E. L., and Peterson, F. E., "SAP IV - A Structural Analysis Program for Static and Dynamic Response of Linear Systems", Report No. EERC 73-11, Earthquake Engineering Research Center, University of California - Berkeley, June, 1973.

4-4 SCL-DR-65-34, "SHOCK - A Computer Code for Solving Lumped Mass Dynamic Systems", V. K. Fabrielson, January, 1966.

4-5 "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions with Application of Analysis to CE 800 Mwe Class Reactors," Combustion Engineering, Inc., Report CENPD-42, August 1972 (Proprietary).

4-6 J. S. Anderson, "Boraflex Neutron Shielding Material -- Product Performance Data," Brand Industries, Inc., Report 748-30-2 (August 1981).

4-7 J. S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1 (August 1981).

4-8 J. S. Anderson, "A Final Report on the Effects of High Temperature Borated Water Exposure on BISCO Boraflex Neutron Absorbing Materials," Branch Industries, Inc., Report 748-21-1 (August 1978).

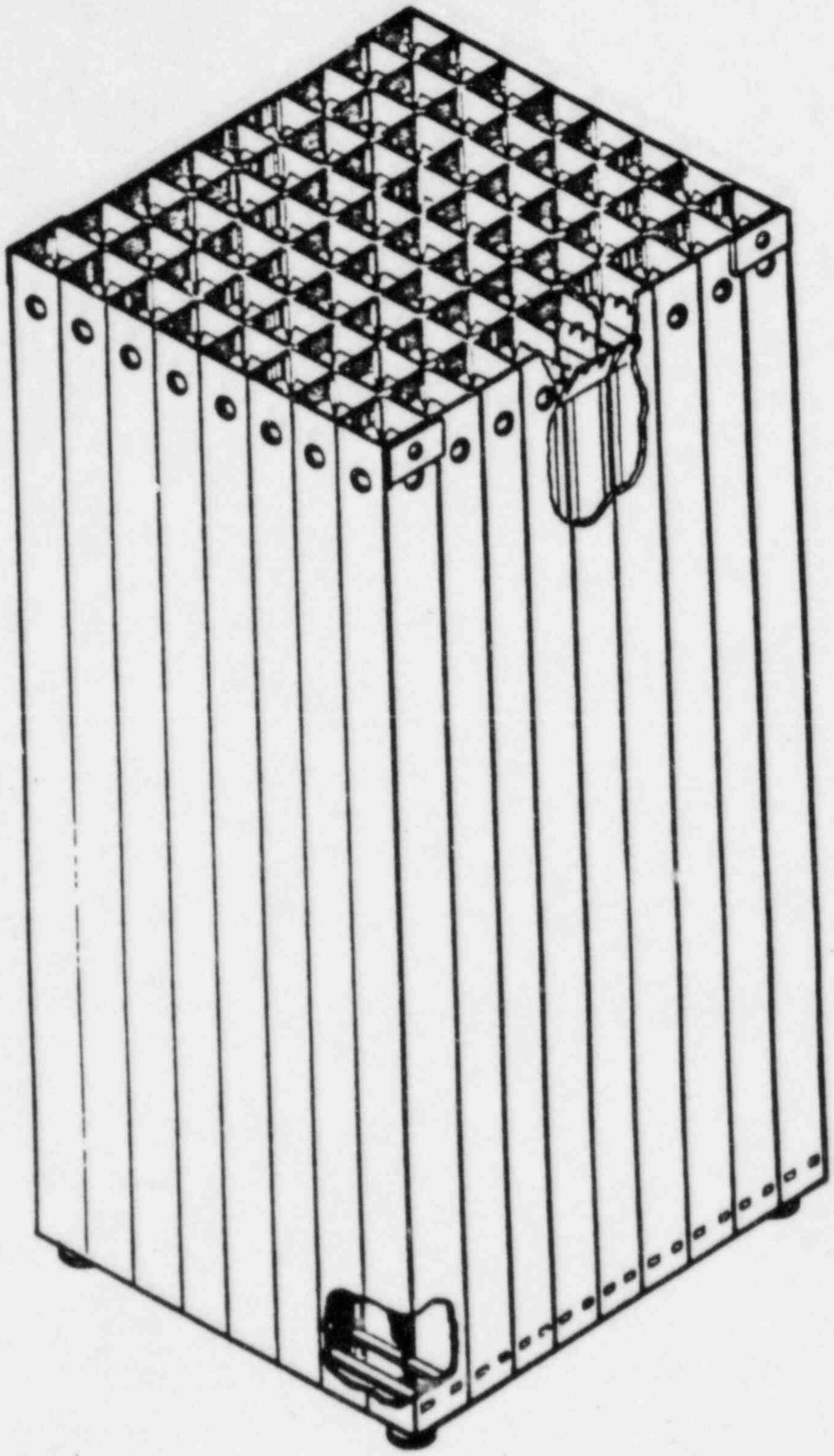

FIGURE 4-1

Typical Spent Fuel Rack Module/
Poison Box
(Region I)

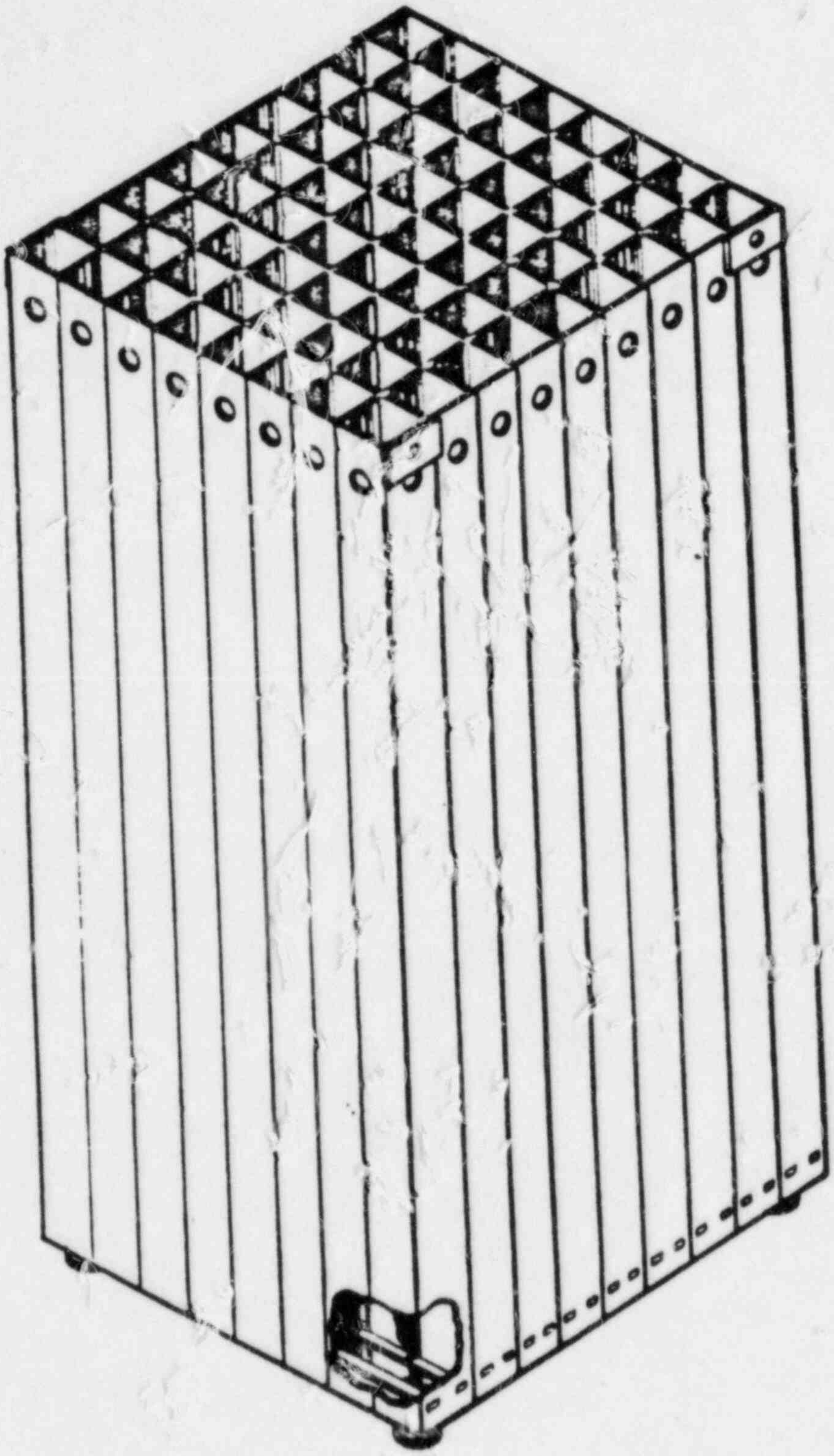

FIGURE 4-2

Typical Spent Fuel Rack Module
(Region II)

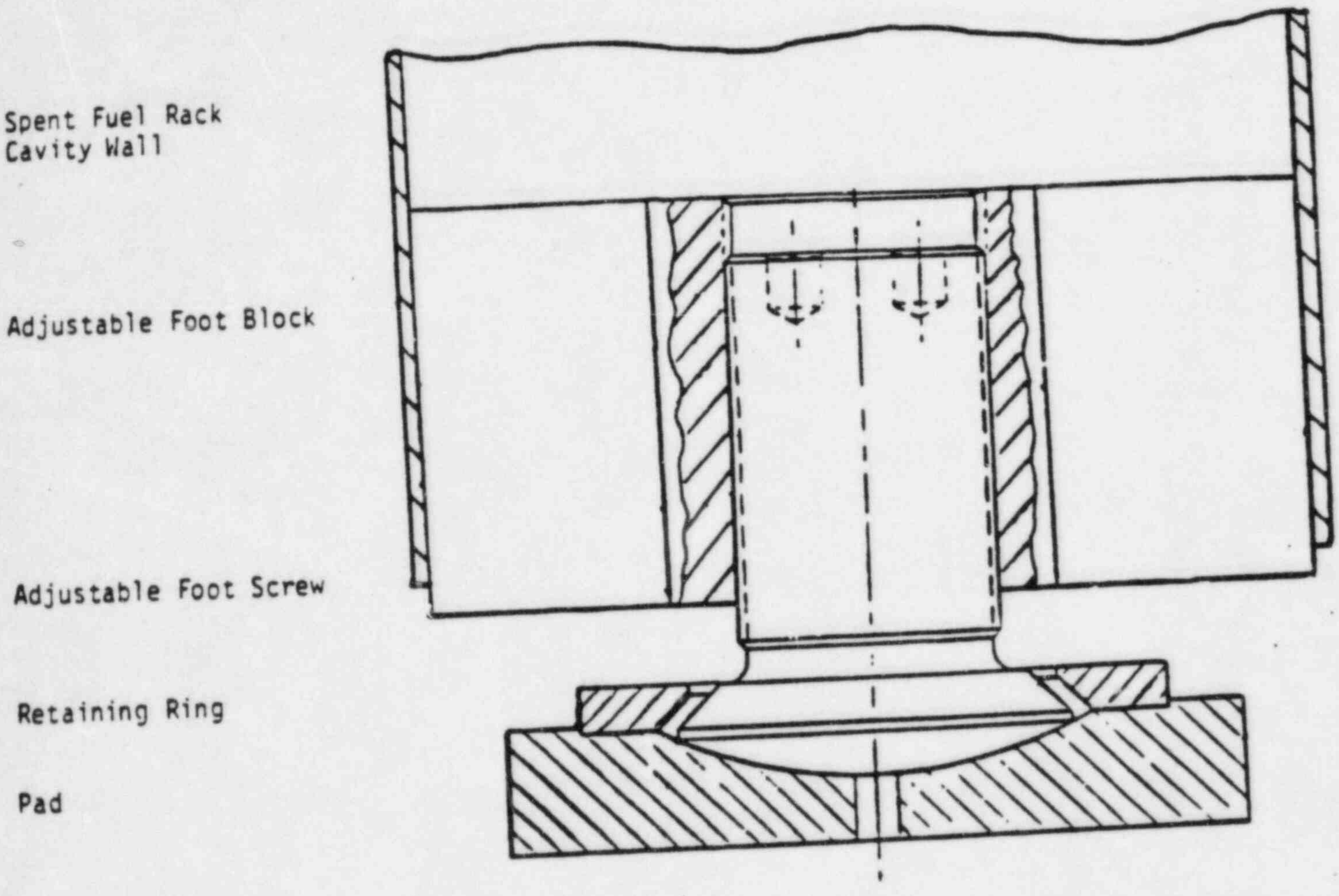

FIGURE 4-3

Adjustable Foot

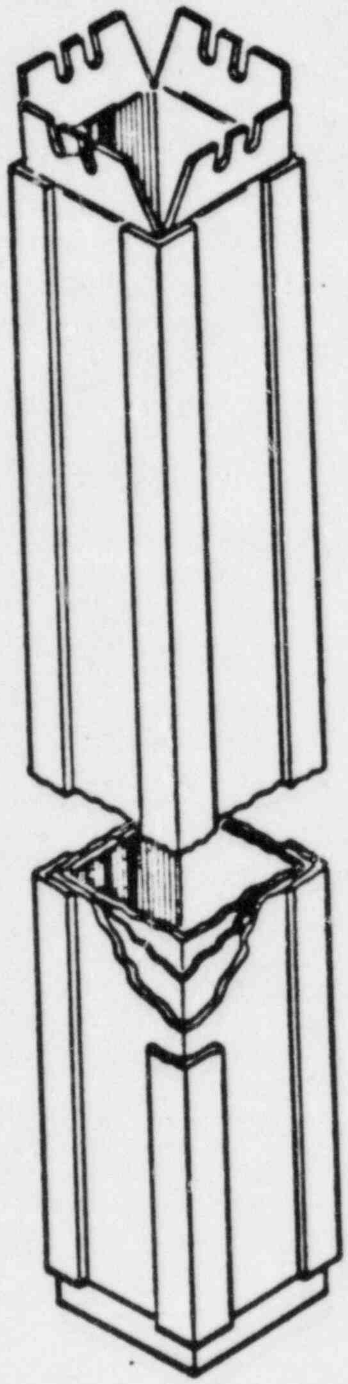

FIGURE 4-4

Poison Box

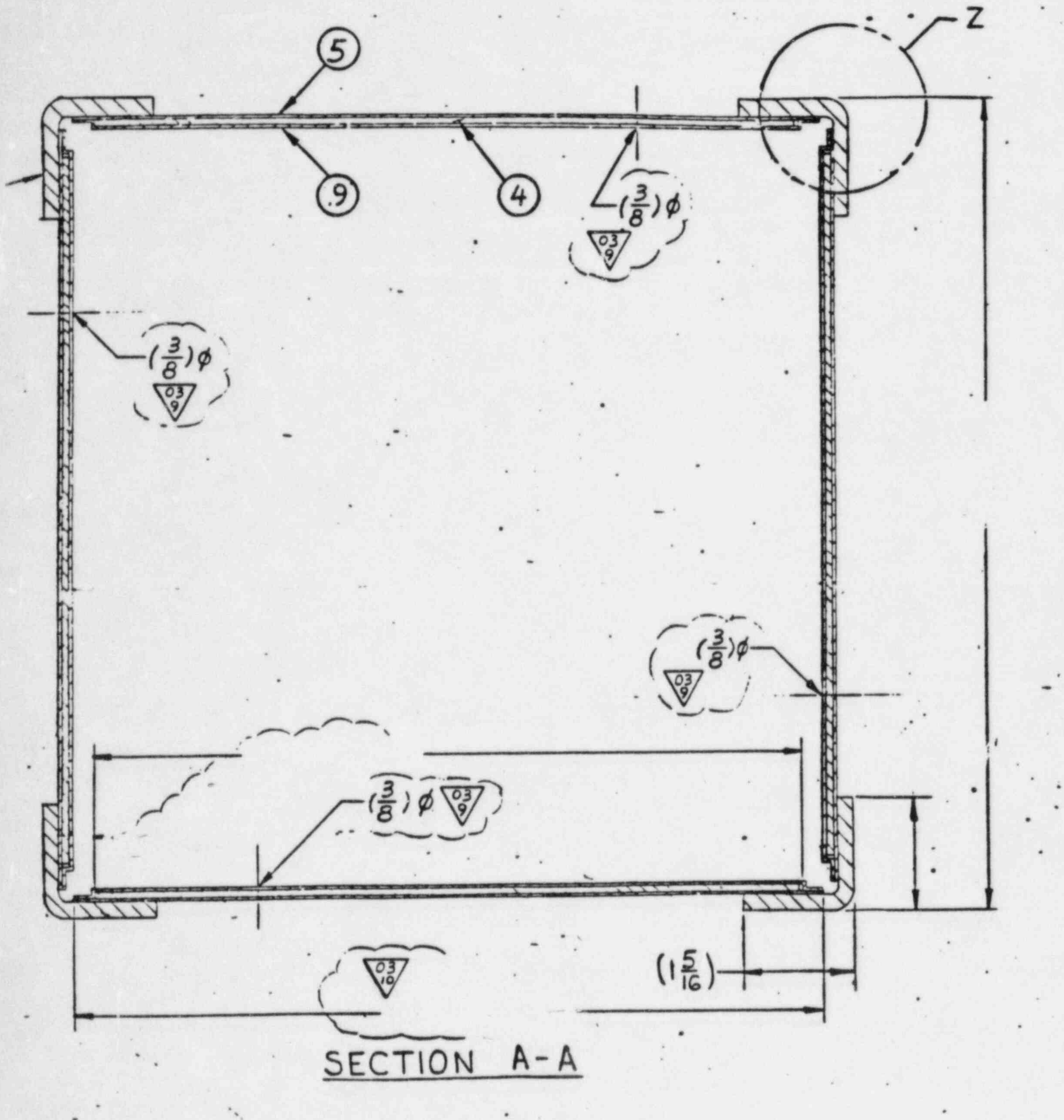

FIGURE 4-5

Poison Box Section View

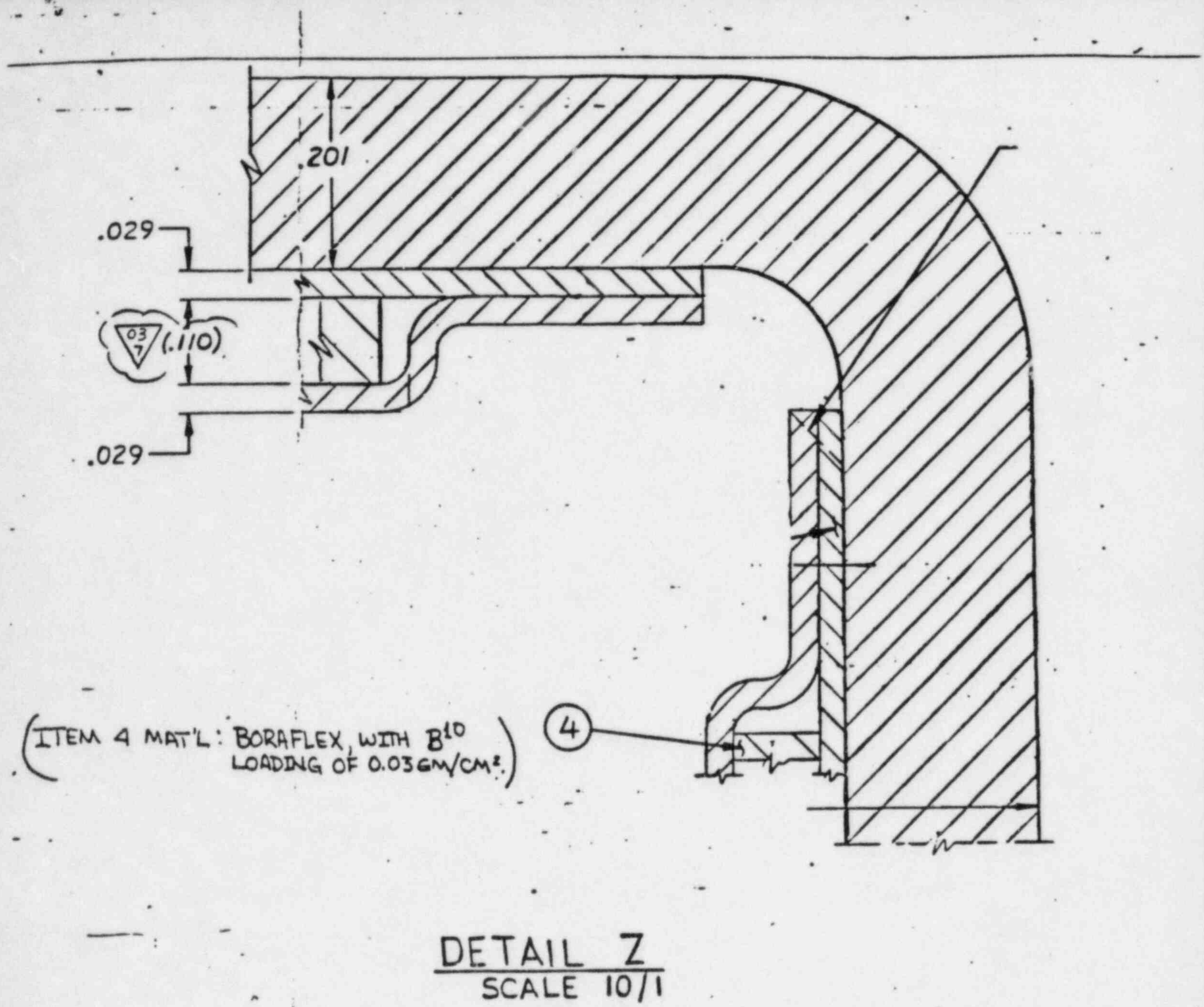

FIGURE 4-6

Poison Box Section Detail

5.0 COST ASSESSMENT AND ENVIRONMENTAL IMPACT

5.1 COST ASSESSMENT

5.1.1 Need For Increased Storage Capacity

A. NNECO currently has no contractual arrangements with any fuel reprocessing facilities.

B. Adoption of this proposed spent fuel storage expansion would not necessarily extend the time period that spent fuel assemblies would be stored on site. Spent fuel could be sent off site for final disposition under existing legislation, but the government facility is not expected to be available before 1998. As matters now stand and until alternate storage facilities are available, spent fuel assemblies on site will remain there.

C. It is estimated that the spent fuel pool will be filled, with the proposed increase in storage capacity, in 1993.

5.1.2 Construction Costs

Total construction cost associated with the proposed modification is approximately 7.1 million dollars. This figure includes the cost of designing and fabricating the spent fuel racks, engineering costs, and installation and support costs at the site.

5.2 ENVIRONMENTAL EFFECTS

5.2.1 Heat Dissipation Effects

This section evaluates the changes in thermal effects due to the proposed increased spent fuel storage capacity. Because spent fuel assemblies will be added periodically, and it is not anticipated that any will be shipped off-site in the foreseeable future, the annual average heat loads in the spent fuel pool will increase as more assemblies are added to the spent fuel pool.

As indicated in Section 3.2, the calculated increase in heat load for a normal refueling sequence is 3.9×10^6 BTU/hr. This would result in an increase in the spent fuel pool water temperature of 11°F, from 120°F to 131°F. This added heat load would cause a minimal temperature rise in the once-through cooling system discharge water temperature.

5.2.2 Radiological Considerations

5.2.2.1 Normal Operations

The radiological consequences of increasing the capacity of the Millstone Unit No. 2 spent fuel pool from 301 fuel assemblies to 667 assemblies were previously analyzed in Reference 5-1 (Sec. 7.0). The results of this

analysis have been extrapolated to predict the radiological impact of increasing the spent fuel pool capacity from 667 to 1112 assemblies. The predicted dose rates from 667 assemblies were:

Location	Dose Rate (mRem/hr)
Refueling Platform	2.54
Poolside	1.54

Extrapolating by a ratio of 1112/667 (1.7) yields the new dose rates for 1,112 assemblies:

Location	Dose Rate (mRem/hr)
Refueling Platform	4.23
Poolside	2.57

Extrapolation in this fashion will yield conservatively high dose rates because the original model used to predict dose rates was based primarily on leaky fuel. Leaving additional old spent fuel assemblies in the pool will not increase the source term linearly as the above ratio implies because decay will result in some of the short lived activity decreasing and, according to Reference 5-1, the cobalt source activity (approximately 75% of the dose rate) is independent of fuel loading.

The incremental increase in cumulative dose based on the Reference 5-1 occupancy times is:

Refueling platform

(4.23 mR/hr - 2.54 mR/hr) (400 Man-hrs) = 0.676 Man Rem

Poolside

(2.57 mR/hr - 1.54 mR/hr) (200 Man-hrs) = 0.206 Man Rem

TOTAL 0.882 Man Rem

Thus, the modification will result in an increase in cumulative dose that is conservatively estimated to be below 1 Man Rem.

5.2.2.2 Accident Conditions

Similar to Reference 5-1, the proposed modification does not affect the basis for the safety analysis of the fuel handling accident discussed in Section 14.19 of the Millstone Unit No. 2 FSAR. As such, the results of that analysis are still applicable.

In Reference 5-2 the requirement for a specified decay time of 120 days for fuel stored within a distance L from the center of the spent fuel pool cask set-down area was identified. The distance L equals the major dimension of the shielded cask. While the proposed modification does not affect the basis for the safety analysis of the cask drop accident, review of the consequences of a cask drop accident with respect to the increased fuel storage density determined that the site boundary two hour whole body dose would increase by 100 millirem. While the new calculated value for the site boundary two hour whole body dose of 240 millirem is well within the requirements established in 10CFR100 it has been determined to be an unreviewed safety question pursuant to 10CFR50.59(a)(2)(i). The actual increase in consequences is acceptable from a safety standpoint.

5.2.3 Chemical Discharges

The water from the Millstone Unit No. 2 spent fuel pool is purified by passing it through the spent fuel pool cleanup system and returning it to the pool. This cleanup system consists of filters and a demineralizer to remove radioactive nuclides and chemical impurities in the water. The wastes generated by the system consist of a small volume of filter cartridges and resins which are packaged and shipped offsite as solids to an approved burial site. There is no change expected in the environmental effects from chemical discharges over that previously evaluated.

5.3 REFERENCES

5-1 Letter, D. C. Switzer (NNECO) to G. Lear (NRC) under Docket No. 50-336 dated November 22, 1976, "Modifications to Spent Fuel Storage Pool".

5-2 Letter, G. Lear (NRC) to D. C. Switzer (NNECO) dated June 30, 1977.

Attachment 3

Millstone Unit No. 2
No Significant Hazards
Consideration Determination

July, 1985

I. Introduction

10CFR50.91 requires that requests for an amendment to an operating license be accompanied by an analysis "about the issue of no significant hazards consideration." Such analysis is to focus on the three standards set forth in 10CFR50.91(b) as quoted below:

> The Commission may make a final determination pursuant to the procedures in 50.91 that a proposed amendment to an operating license for a facility licensed under 50.91(b) or 50.22 or for a testing facility involves no significant hazards consideration, unless it finds that operation of the facility in accordance with the proposed amendment would:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

Northeast Nuclear Energy Company (NNECO) submits that the activities associated with reracking the Millstone Unit No. 2 spent fuel pool are outside the standards set forth in 10CFR50.91(b) and, accordingly, a no significant hazards consideration finding is warranted with respect to the proposed license amendment.

II. Background

Millstone Unit No. 2 was designed and constructed with a spent fuel pool with a capacity of 301 spent fuel assemblies. The Millstone Unit No. 2 Final Safety Analysis Report addresses the safety and environmental implications of the pool, including the relevant acceptance criteria with respect to criticality, structural integrity, and cooling. The NRC found the environmental and safety impacts of spent fuel storage to be acceptable.

By November 1976 it became necessary for NNECO to provide greater onsite capacity for storage of spent fuel. Backend fuel cycle services, particularly spent fuel reprocessing, had not materialized as originally anticipated. A spent fuel pool rerack amendment was approved and issued by the NRC on June 30, 1977. The reracking amendment allowed modifications to increase pool storage capacity to 667 fuel assemblies.

Following the Cycle 6 refueling outage in 1985, it is again necessary for NNECO to increase the capacity of the Millstone Unit No. 2 spent fuel pool to provide onsite capacity for storage of spent fuel. The proposed reracking project will increase capacity to 1112 fuel assemblies and is described more fully in the accompanying "Millstone Nuclear Power Station, Unit No. 2, Spent Fuel Rerack Safety Analysis Report" (Attachment 2). In sum, NNECO will utilize a region strategy, including regions of both poisoned and non-poisoned spent fuel racks, and in one region taking credit for reactivity depletion in spent fuel to provide substantially closer center-to-center spacing of fuel assemblies for

increased capacities. The racks have been designed and will be provided by Combustion Engineering, Inc. (CE). CE racks of this type have been most recently licensed by the NRC for use at Florida Power and Light Company's St. Lucie and at Arizona Public Service's Palo Verde nuclear plants.

The following no significant hazards evaluation relies upon the analysis contained in Attachment 2 to demonstrate that none of the three standards defining a matter involving significant hazards considerations is met. Each of the three standards is discussed below.

III. Application of Standards

A. First Standard

> Involve a significant increase in the probability or consequence of an accident previously evaluated.

NNECO's safety analysis of the proposed reracking has been accomplished using current NRC Staff accepted Codes and Standards as specified in Section 4.2 of Attachment 2. The results of the safety analysis demonstrate that the proposal meets the specified acceptance criteria set forth in these standards. In addition, NNECO has reviewed NRC Staff Safety Evaluation Reports for prior spent fuel pool rerackings involving spent fuel pool rack replacements to ensure that there are no identified concerns not fully addressed in this submittal. NNECO has identified no such concerns.

From our analysis and reviews, NNECO has identified the following potential accident scenarios: 1) spent fuel cask drop; 2) loss of spent fuel pool forced cooling; 3) seismic event; 4) spent fuel assembly drop; 5) criticality accident; and 6) Load Handling Accident. The probability of the occurrence of any of the first four listed accidents is not affected by the racks themselves; thus, reracking cannot increase the probability of these accidents.

All potential events which could involve accidental criticality have been examined in Section 3.1.2 of Attachment 2. It was concluded

that the bounding accident was dropping an unirradiated fuel assembly into a blocked fourth location in Region II. The probability of dropping a fuel assembly during fuel movement operations is not affected by the fuel storage racks.

The proposed Millstone Unit No. 2 spent fuel pool reracking will not involve an increase in probability of any previously evaluated load handling accident as accepted standards and procedures will be utilized as described in Section 3.3.2 of Attachment 2.

The consequences of the (1) spent fuel cask drop accident have been evaluated as described in Section 5.4 and 9.8 of the Millstone Unit No. 2 Final Safety Analysis Report (FSAR). By controlling the decay time for fuel stored within a specified distance from the cask set down area to not less than 120 days prior to cask movement together with an administrative control specifying a minimum required boron concentration in the water of the spent fuel pool, the consequences of this accident type will remain well within 10CFR100 guidelines.

There is, however, an increase in the value of the two hour whole body dose at the site exclusion boundary for a postulated cask drop accident. The new racks increase the storage density of spent fuel within the distance L of the cask set-down area. This results in a calculated increase of the two hour whole body dose from 140 millirem to 240 millirem, an increase of 100 millirem. In review of this submittal, NNECO has recognized this increase and has designated it an unreviewed safety question. The calculated dose is well within the guidelines specified by 10CFR100 and as such the consequences of this type of accident will not be significantly increased from previously evaluated events.

The consequences of the (2) loss of spent fuel pool forced cooling accident have been evaluated and are described in Section 3.2 of Attachment 2. As indicated in Table 3-2 there is ample time to effect repairs of the cooling system or to establish makeup flow to the spent fuel pool. The consequences of this type accident will not be significantly increased from previously evaluated accidents by this proposed reracking.

The consequences of a (3) seismic event have been evaluated and are described in Sections 4.3 and 4.5 of Attachment 2. The racks were evaluated against the appropriate NRC standards as described in Section 4.2 of Attachment 2. The results of the seismic and structural analysis show that the proposed racks meet all of the NRC structural acceptance criteria and are consistent with results found acceptable by the NRC Staff in previous poison rerack SERs. Thus, the consequences of seismic events will not significantly increase from previously evaluated seismic events.

The consequences of a (4) spent fuel assembly drop accident are described in Section 14.19 of the Millstone Unit No. 2 FSAR. A complete list of assumptions is provided in FSAR Table 14.19-1. Results of the analysis are well below the limits of 10CFR100 and are presented in Section 14.19.3. The consequences of this type accident will not be significantly increased from previously evaluated accidents by this proposed reracking.

The consequences of a (5) criticality accident have been evaluated in Section 3.1.2 of Attachment 2. All potential events which could involve accidental criticality have been examined. The bounding criticality accident was found to be the dropping of a fresh fuel assembly into a blocked fourth location in Region II. Administrative controls in the form of a Technical Specification of minimum boron concentration for the water of the spent fuel pool (Attachment 1) will preclude the bounding criticality accident; therefore, the consequences of this type accident will not be significantly increased from previous accident evaluations by this proposed reracking.

The consequences of a (6) load handling accident have been evaluated in Section 3.3.2 of Attachment 2. The work to be done in the spent fuel pool will be performed in accordance with accepted construction practices, standards, and procedures. The consequences of this type accident will not be significantly increased from previous accident evaluations by this proposed reracking. Therefore, it is shown that the proposed Millstone Unit No. 2 spent fuel rack replacement will not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. Second Standard

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

NNECO has evaluated the proposed rack replacement in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", appropriate NRC Regulatory Guides, appropriate NRC Standard Review Plan sections, and appropriate industry Codes and Standards as described in Section 4.2 of Attachment 2. In addition, NNECO has reviewed the NRC Safety Evaluation Report for the previous Millstone Unit No. 2 spent fuel rack replacement application and for other prior spent fuel pool rerackings.

The change to a two-region spent fuel pool creates the requirement to perform additional evaluations to ensure the criticality requirement is maintained. These include the evaluation of the limiting condition (dropping a fresh fuel assembly into a blocked fourth location in Region II). This evaluation shows that, when the boron concentration requirement is met, per the proposed Technical Specifications, the criticality criterion is satisfied. Although this change does create the requirement to address additional aspects of a previously analyzed accident, it does not create the possibility of a previously unanalyzed accident.

C. Third Standard

Involve a significant reduction in a margin of safety.

The issue of "margin of safety," when applied to a spent fuel rack replacement, includes the following considerations:

a. Nuclear criticality considerations.

b. Thermal hydraulic considerations.

c. Mechanical, material, and structural considerations.

The margin of safety that has been established for nuclear criticality is that the neutron multiplication factor (K_{eff}) in the spent fuel pool is to be $\leq$ 0.95, including all uncertainties, under all conditions. For the proposed modification, the criticality analysis is described in Section 3.1 of the SAR. The methods utilized in the analysis conform with ANSI N210-1976, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations"; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety"; the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (April 1978), as modified (January 1979); and Regulatory Guide 1.13, "Spent Fuel Facility Design Basis," proposed Revision 2. The computer programs,

data libraries, and benchmarking data used in the evaluation have been used in previous spent fuel rack replacement applications by other NRC licensees and have been reviewed and approved by the NRC. The results of NNECO's analysis indicate that K_{eff} is ≤ 0.95 under all postulated conditions, including uncertainties, at a 95/95 probability/confidence level. Thus, meeting the acceptance criteria for criticality, the proposed reracking does not involve a significant reduction in the margin of safety for nuclear criticality.

For thermal hydraulics, the relevant considerations for evaluating if there is a significant reduction in margin of safety are: (1) maximum fuel temperature, and (2) the increase in temperature of the water in the pool. NNECO's thermal hydraulic evaluation is described in Section 3.2 of Attachment 2. Results of this analysis show that fuel cladding temperatures under abnormal conditions are sufficiently low to preclude structural failure and that boiling does not occur in the water channels between the fuel assemblies nor within the storage cells. However, the proposed rack replacement will result in an increase in the maximum heat load in the Millstone Unit No. 2 spent fuel pool. As shown in Section 3.2, the maximum temperature will not exceed the current margin of safety (150°F). For the maximum normal heat load case (full-core discharge at 150 hr. after shutdown, which fills the spent fuel pool to its capacity), the pool temperature will not exceed 150°F. Thus, there is no significant reduction in the margin of safety from a thermal hydraulic standpoint or from a spent fuel pool cooling standpoint.

The mechanical, material, and structural considerations of the proposed rack replacement are analyzed in Section 4.0 of Attachment 2. As described in Section 4.2, the racks are designed in accordance with the applicable NRC Regulatory Guides, Standard Review Plan sections, and position papers, as well as the appropriate industry Codes and Standards. The racks are designed to Seismic Category I requirements. The materials utilized are described in Section 4.6.1 and are compatible with the spent fuel pool and the spent fuel assemblies. The structural considerations are also described in Section 4.0 of Attachment 2. The conclusion of the analysis is that the margin of safety is not significantly reduced by the proposed reracking.

IV. Conclusion

The proposed Millstone Unit No. 2 reracking does not:

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

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

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

As such, NNECO has determined and submits that the proposed spent fuel pool reracking does not involve any significant hazards consideration.

Docket No. 50-336
B11549

Attachment 4

Millstone Nuclear Power Station, Unit No. 2

Questions Identified by NRC
Staff During May 17, 1984
Bethesda Meeting

July, 1985

In a May 17, 1984 meeting in Bethesda between Northeast Nuclear Energy Company (NNECO) and the NRC, the Staff identified four items of the rerack proposal which would require additional information.

Each of these items is addressed below:

- o Provide details on the thickness of material on either side of the Boraflex.

  Figure 4-6 of the Attachment 2 provides a detail of the inner and outer walls of the poison box.

- o Describe the method to be used to vent the Boraflex (the poison).

  As shown in the Section view, Figure 4-5 of the Attachment, a 3/8" diameter vent hole is provided in the inner wall of the poison box. This vent hole, in addition to the fact that the walls enclosing the Boraflex are spot welded not seal welded, will allow for proper venting of the poison.

- o Provide details of the surveillance program for materials in the spent fuel pool.

  Section 4.7 of the Attachment 2 deals with the proposed surveillance and testing program.

- o List all materials in the spent fuel pool, especially any plastics or heat treated parts.

  Section 4.6.1 of the Attachment 2 lists the materials used in construction of the spent fuel storage racks and poison boxes.