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Nuclear Energy

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The Northeast Utilities System

89

MAR 1 9 1999
Docket No. 50-423
B17343

Re: 10CFR2.790
10CFR50.90

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

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specification
Spent Fuel Pool Rerack (TSCR 3-22-98)

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed revision into Millstone Unit No. 3 Technical Specifications.

Pursuant to 10 CFR 2.790, it is requested that the proprietary version of the "Licensing Report for Spent Fuel Rack Installation at Millstone Unit 3," (see Attachment 6) discussing the analysis methodology utilized, be withheld from public disclosure. Upon separation of the proprietary version of the "Licensing Report for Spent Fuel Rack Installation at Millstone Nuclear Station Unit 3," from this letter, this letter may be decontrolled.

Description of Proposed Revision

Millstone Unit No. 3 must rerack its spent fuel pool to maintain full core reserve capability approaching the end of its operating license. NNECO proposes to achieve this goal by installing two types of additional higher density spent fuel racks into the spent fuel pool. Existing spent fuel racks will remain in the pool, but are reanalyzed to only accept fuel lower in reactivity than they are licensed to accept at present. The proposed additional racks will have a closer assembly to assembly spacing to increase fuel storage capacity.

Markup of Proposed Revision

A copy of the marked up Technical Specification pages are contained in Attachment 1. The marked up pages reflect the currently issued version of the Technical Specifications. Pending Technical Specification revisions are not reflected in the enclosed markup.

Retype of Proposed Revision

A copy of the retyped Technical Specification pages are contained in Attachment 2. The retyped pages reflect the incorporation of the proposed changes into the currently issued version of the Technical Specifications. Pending Technical Specification revisions are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with recently issued Technical Specifications prior to issuance.

Background, Safety Summary, Significant Hazards Consideration and Environmental Considerations

The Background, Safety Summary, Significant Hazards Consideration and Environmental Considerations that support this proposed revision are contained in Attachments 3 and 4.

Licensing Report

The Licensing Report that supports this proposed revision is contained in Attachments 5 and 6. A copy of a non-proprietary version of the "Licensing Report for Spent Fuel Rack Installation at Millstone Nuclear Station Unit 3," discussing the analysis methodology utilized is enclosed as Attachment 5.

A copy of the proprietary version of the "Licensing Report for Spent Fuel Rack Installation at Millstone Nuclear Station Unit 3," discussing the analysis methodology utilized is enclosed as Attachment 6 (only in the USNRC Document Control Desk copy). Pursuant to 10 CFR 2.790, it is requested that this document be withheld from public disclosure. Upon separation of the proprietary version of the "Licensing Report for Spent Fuel Rack Installation at Millstone Nuclear Station Unit 3," from this letter, this letter may be decontrolled.

Affidavit Pursuant to 10 CFR 2.790 for Control of Proprietary Holtec Report No.: HI-971843

An affidavit pursuant to 10 CFR 2.790 is enclosed in Attachment 6.

Plant Operations Review Committee and Nuclear Safety Assessment Board Review

The Plant Operations Review Committee and the Nuclear Safety Assessment Board have reviewed this proposed amendment request and concur with the contained determinations.

State Notification

In accordance with 10CFR50.91(b), we are providing the State of Connecticut with a copy of this proposed amendment to ensure their awareness of this request.

Schedule Request for NRC Approval

NNECO requests NRC review and approval of this proposed revision by June 2000. Additionally, NNECO's January 18, 1999 (B17004), submittal regarding Full Core Offload capability, contains an assumed heat load which bounds the heat load associated with this rerack licensing amendment request and NRC approval of the January 18, 1999, submittal is required prior to approval of this rerack licensing amendment. NNECO also requests that the NRC allow implementation of the approved revision per the special circumstance regarding transitioning to the revised technical specifications described in Attachment 3.

There are no regulatory commitments contained within this letter.

If the NRC Staff should have any questions or comments regarding this submittal, please contact Mr. D. Dodson at (860) 447-1791 ext. 2346.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

R. P. Necci
Vice President - Nuclear Oversight and
Regulatory Affairs

Subscribed and sworn to before me

this 19 day of March, 1999

Loretta F. Goodson

Date Commission Expires: LORETTA F. GOODSON
NOTARY PUBLIC
Commission Expires November 30, 2001

cc: See page 4

cc: H. J. Miller, Region I Administrator
J. W. Andersen, NRC Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

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

Docket No. 50-423
B17343

Attachment 1

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specification
Spent Fuel Pool Rerack (TSCR 3-22-98)

Marked Up Pages

March 1999

MARKUP OF PROPOSED REVISION

Refer to the attached markup of the proposed revision to the Technical Specifications. The attached marked up pages reflect the currently issued version of the Technical Specifications listed below. Pending Technical Specification revisions or Technical Specification revisions issued subsequent to this submittal are not reflected in the enclosed markup.

The proposed changes to the Millstone Unit No. 3 Technical Specifications are summarized below, and are attached

Revise INDEX pages xii and xv for new figures and page numbers.

Revise Section 1.40: SPENT FUEL POOL STORAGE PATTERN, as follows: Defines the fuel storage pattern based on blocked storage locations, and the associated adjacent and diagonal locations.

Revise Section 1.41: 3-OUT-OF 4 AND 4-OUT-OF-4, as follows: Defines the two possible storage configurations in Region 1 fuel racks.

Revise Section 3.9.1.2: BORON CONCENTRATION, as follows: Section 3.9.1.2 to require that soluble boron concentration be greater than or equal to 800 ppm. Add that the applicability of Section 3.9.1.2 is required during fuel assembly movement in the spent fuel pool, and the action to take if Section 3.9.1.2 is out of specification. Section 4.9.1.2 under SURVEILLANCE REQUIREMENTS requires that boron concentration be verified greater than or equal to 800 ppm prior to fuel assembly movement into or within the spent fuel pool, and every 7 days thereafter during fuel movement.

Revise Section 3/4.9.7: CRANE TRAVEL - SPENT FUEL STORAGE AREAS, as follows: In Section 4.9.7, under SURVEILLANCE REQUIREMENTS, the proposed Technical Specification clarifies that the crane interlocks and stops prevent a crane from carrying a load in excess of 2,200 lbs over the spent fuel pool verses being carried over fuel assemblies as stated in the existing surveillance. The proposed Technical Specification allows fuel pool gates, spent fuel racks or loads less than 2,200 lbs. to be moved by a crane under administrative controls in lieu of crane interlocks and physical stops

Revise Section 3/4.9.13: SPENT FUEL POOL - REACTIVITY as follows: Section 3.9.13b requires that immediate action be initiated to move any misplaced fuel assembly into a location for which the assembly is qualified. Revise Section 4.9.13.1 under SURVEILLANCE REQUIREMENTS, as follows: Renumber Section 4.9.13.1 to 4.9.13.1 which requires appropriate documentation be reviewed to assure that fuel assemblies stored in a 4-out-of-4 storage pattern in Region 1 fuel racks meet the burnup/enrichment requirements of Figure 3.9-1 (replaces old figure). Add Section 4.9.13.1.2 which requires appropriate documentation be reviewed to assure that fuel assemblies stored in Region 2 fuel racks meet the burnup/enrichment requirements of Figure 3.9-3 (new figure). Add Section 4.9.13.1.3 which requires appropriate documentation be reviewed to assure that fuel assemblies stored in Region 3 fuel racks meet the burnup/enrichment/decay time requirements of Figure 3.9-4 (new figure). Delete Sections 3.9.13c and 3.9.13d: SPENT FUEL POOL - REACTIVITY and associated Sections 4.9.13.2 and 4.9.13.3: SURVEILLANCE REQUIREMENTS. These sections deal with Boraflex integrity in response to a dropped load on a fuel rack, and in response to an Operating Basis Earthquake (OBE). The proposed reracking project eliminates all credit for Boraflex, and places burnup/enrichment/decay time requirements on fuel assemblies stored in Boraflex racks to assure k_{eff} remains less than or equal to 0.95.

Revise 3.9.14: SPENT FUEL POOL - STORAGE PATTERN, as follows: Replace the roman numeral I with the number 1 for Region 1 designation. Note, for simplicity and clarity, fuel storage region designation is being changed from roman numerals to standard numbers. This change is editorial in nature, and does not impact the rerack project design or safety.

Replace Figures 3.9-1 and 3.9-2 with new figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 indicating storage requirements for the proposed Regions 1, 2 and 3 fuel racks.

Revise Section 3/4.9.1.1: BASES - BORON CONCENTRATION, as follows: Correct the section designator from 3/4.9.1 to 3/4.9.1.1.

Revise Bases Section 3/4.9.1.2: BASES - BORON CONCENTRATION IN SPENT FUEL POOL, as follows: The proposed Technical Specification Basis no longer mentions Boraflex. The proposed Technical Specification Basis lists the different means by which fuel racks will maintain k_{eff} less than or equal to 0.95. It also states that the boron requirements in Section 3.9.1.2 ensures k_{eff} remains less than or equal to 0.95 for a dropped or misplaced fuel assembly.

Revise Bases Section 3/4.9.13: BASES - SPENT FUEL POOL - REACTIVITY, as follows: The proposed Technical Specification Basis no longer mentions Boraflex. It also lists the different means by which each region of proposed fuel racks will maintain k_{eff} less than or equal to 0.95.

Revise Bases Section 3/4.9.14: BASES - SPENT FUEL POOL - STORAGE PATTERN, as follows: This section is revised to recognize that Region 1 can now be either in a 3-OUT-OF-4, or 4-OUT-OF-4 storage configuration

Revise Section 5.6.1.1: DESIGN FEATURES - CRITICALITY, as follows: This section describes the pitch, neutron absorber, storage pattern, and burnup/enrichment/decay time limits for each region of proposed fuel racks.

Revise Section 5.6.3: DESIGN FEATURES - CAPACITY, as follows: This section lists the storage capacity of each proposed region of fuel racks.

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03/24/94 Pg 1

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.11 RADIOACTIVE EFFLUENTS

2

INSERT A

VENTING

1.39 VENTING shall be 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.

SPENT FUEL POOL STORAGE PATTERNS: Replace with Insert B

1.40 Region I spent fuel racks contain a cell blocking device in every 4th location for criticality control. This 4th location will be referred to as the blocked location. A STORAGE PATTERN refers to the blocked location and all adjacent and diagonal Region I cell locations surrounding the blocked location. Boundary configuration between Region I and Region II must have cell blockers positioned in the outermost row of the Region I perimeter, as shown in Figure 3.9-2.

1.41 Region II contains no cell blockers.

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

ALLOWED POWER LEVEL

1.43 APL^{ND} is the minimum allowable nuclear design power level for base load operation and is specified in the COLR.

1.44 APL^{BL} is the maximum allowable power level when transitioning into base load operation.

INSERT B

5

STORAGE PATTERN

1.40 STORAGE PATTERN refers to the blocked location in a Region 1 fuel storage rack and all adjacent and diagonal Region 1 (or Region 2) cell locations surrounding the blocked location. The blocked location is for criticality control.

3-OUT-OF-4 and 4-OUT-OF-4

1.41 Region 1 spent fuel racks can store fuel in either of 2 ways:

(a) Areas of the Region 1 spent fuel racks with fuel allowed in every storage location are referred to as the 4-OUT-OF-4 Region 1 storage area.

(b) Areas of the Region 1 spent fuel racks which contain a cell blocking device in every 4th location for criticality control, are referred to as the 3-OUT-OF-4 Region 1 storage area. A STORAGE PATTERN is a subset of the 3-OUT-OF-4 Region 1 storage area.

4/9/98 6

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.2 The boron concentration of the Spent Fuel Pool shall be maintained uniform and sufficient to ensure that the boron concentration is greater than or equal to 1750 ppm.

Applicability

Whenever fuel assemblies are in the spent fuel pool.

Action

a. With the boron concentration less than 1750 ppm, initiate action to bring the boron concentration in the fuel pool to at least 1750 ppm within 72 hours, and

b. With the boron concentration less than 1750 ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to 1750 ppm every 72 hours.

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INSERT C

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be maintained uniform, and greater than or equal to 800 ppm.

<u>Applicability</u>
During all fuel assembly movements within the spent fuel pool.

<u>Action</u>
With the spent fuel pool soluble boron concentration less than 800 ppm, suspend the movement of all fuel assemblies within the spent fuel pool.

<u>Surveillance Requirements</u>

4.9.1.2 **Verify that the soluble boron concentration is greater than or equal to 800 ppm prior to any movement of a fuel assembly into or within the spent fuel pool, and every 7 days thereafter during fuel movement**

October 25, 1990

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2200 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. **With the requirements of the above specification not satisfied, place the crane load in a safe condition.**

b. **The provisions of Specification 3.0.3 are not applicable.**

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops ~~which prevent crane travel with~~ loads in excess of 2200 pounds over the fuel storage pool ~~fuel assemblies~~ shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation. Administrative controls may be used in lieu of crane interlocks and physical stops for handling fuel racks, spent fuel pool gates, or loads less than 2200 pounds.

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3/4.9.13 SPENT FUEL POOL - REACTIVITY

LIMITING CONDITION FOR OPERATION

3.9.13 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 assemblies are in the spent fuel pool.

ACTION: With k_{eff} greater than 0.95:

a. Borate the Spent Fuel Pool until k_{eff} is less than or equal to 0.95, and

b. Initiate action to correct the cause of the misplaced/dropped fuel assembly, if required, and

Replace w/ INSERT E

~~c. Following the drop of a load on the spent fuel racks, with fuel in the fuel rack location, close and administratively control the opening of dilution pathways to the Spent Fuel Pool until Boraflex in the Spent Fuel Pool is determined to be within design, and~~

~~d. Following a seismic event of Operating Basis Earthquake (OBE) magnitude or greater:~~

~~1) Close and administratively control the opening of dilution pathways to the Spent Fuel Pool until Boraflex in the Spent Fuel Pool is determined to be within design.~~

~~2) Notify the Commission of the action taken for Spent Fuel Reactivity control as part of the report required by Specification 4.3.3.3.2.~~

ELIMINATE

SURVEILLANCE REQUIREMENTS

Replace w/ INSERT D

4.9.13.1 Ensure that all fuel assemblies to be placed in Region II of the spent fuel pool are within the enrichment and burn-up limits of Figure 3.9-1 by checking the fuel assembly's design and burn-up documentation.

ELIMINATE

~~4.9.13.2 Following a seismic event of Operating Basis Earthquake (OBE) magnitude or greater, perform an engineering evaluation to determine that k_{eff} is less than or equal to 0.95 and that soluble boron is not required for control of k_{eff} in the Spent Fuel Pool. Pending completion of engineering evaluation, take action as required for k_{eff} being greater than 0.95.~~

~~4.9.13.3 Following the drop of a load on the Spent Fuel Racks, with fuel in the fuel rack location, perform an engineering evaluation to determine that k_{eff} is less than or equal to 0.95 and that soluble boron is not required for control of k_{eff} in the Spent Fuel Pool. Pending completion of engineering evaluation, take action as required for k_{eff} being greater than 0.95.~~

INSERT D

4.9.13.1~~a~~.1 Ensure that all fuel assemblies to be placed in Region 1 "4-OUT-OF-4" fuel storage are within the enrichment and burnup limits of Figure 3.9-1 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1~~b~~.2 Ensure that all fuel assemblies to be placed in Region 2 fuel storage are within the enrichment and burnup limits of Figure 3.9-3 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1~~c~~.3 Ensure that all fuel assemblies to be placed in Region 3 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-4 by checking the fuel assembly's design, decay time, and burn-up documentation.

INSERT E

b. Initiate immediate action to move any fuel assembly which does not meet the requirements of Figures 3.9-1, 3.9-3 or 3.9-4, to a location for which that fuel assembly is allowed.

12a

August 29, 1989

REFUELING OPERATIONS

SPENT FUEL POOL - STORAGE PATTERN

LIMITING CONDITION FOR OPERATION

1

3.9.14 Each STORAGE PATTERN of the Region X spent fuel pool racks shall require that:

a. Prior to storing fuel assemblies in the STORAGE PATTERN per Figure 3.9-2, the cell blocking device for the cell location must be installed.

b. Prior to removal of a cell blocking device from the cell location per Figure 3.9-2, the STORAGE PATTERN must be vacant of all stored fuel assemblies.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION: Take immediate action to comply with 3.9.14(a), (b).

SURVEILLANCE REQUIREMENT

4.9.14 Verify that 3.9.14 is satisfied with no fuel assemblies stored in the STORAGE PATTERN prior to installing and removing a cell blocking device in the spent fuel racks.

13

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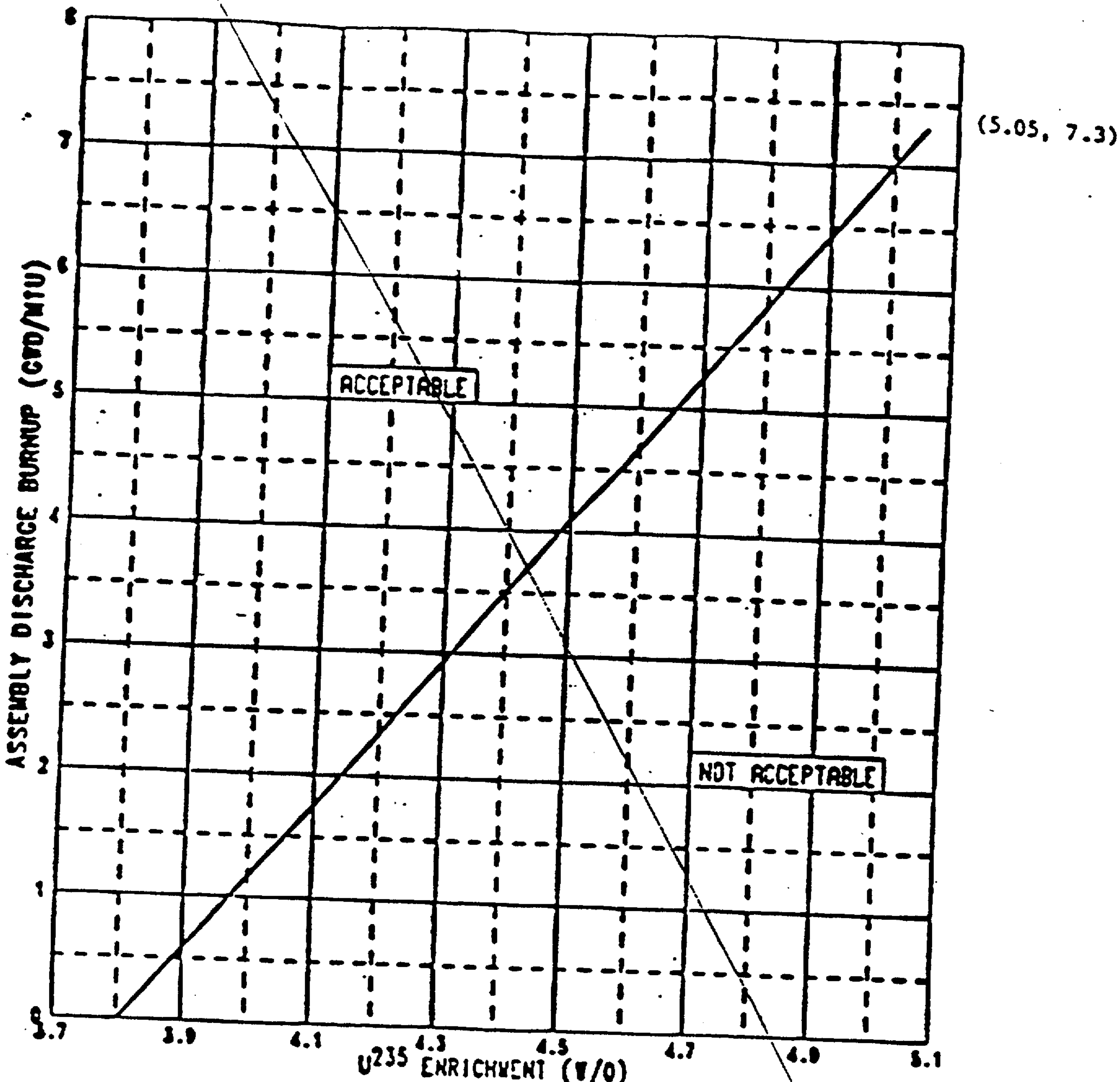
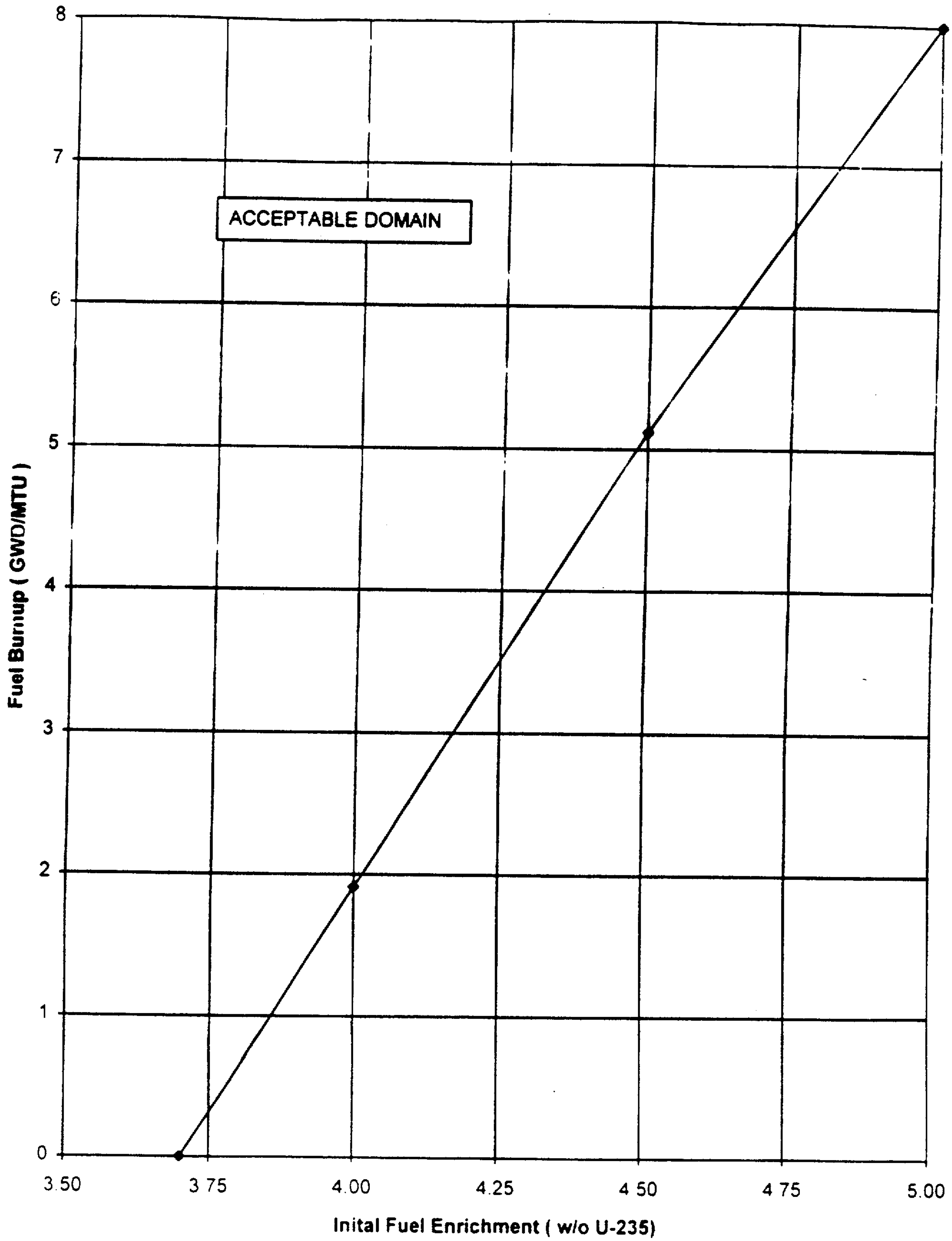


Figure 3.9-1

MILLSTONE UNIT 3 FUEL ASSEMBLY MINIMUM BURNUP VS INITIAL U235 ENRICHMENT FOR STORAGE IN REGION II SPENT FUEL RACKS

FIGURE 3.9-1 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 1 4-OUT-OF-4 Fuel Storage Configuration



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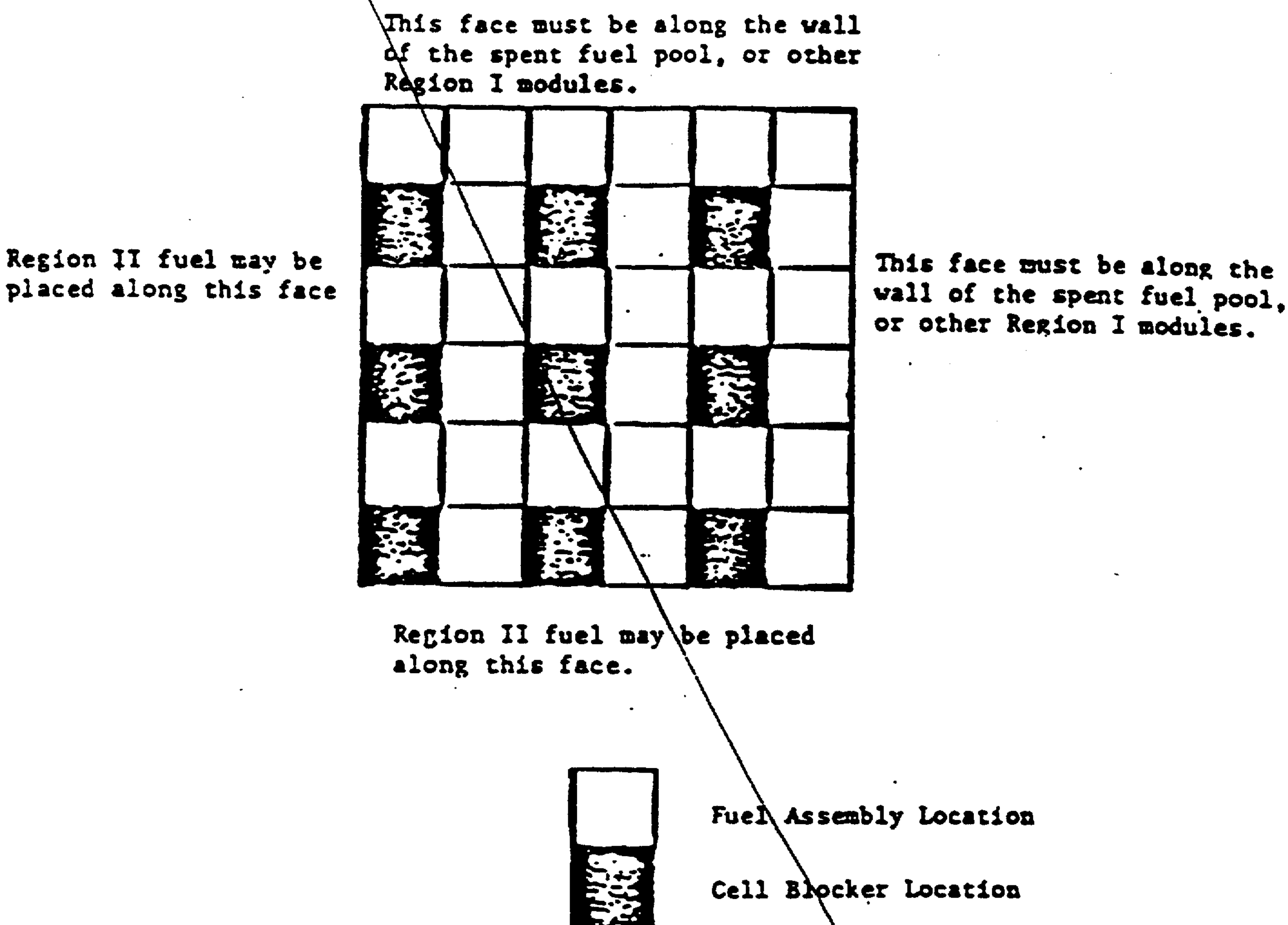

Figure 3.9-2

MILLSTONE UNIT 3 REGION I THREE OF FOUR FUEL ASSEMBLY LOADING SCHEMATIC FOR A TYPICAL 6 X 6 STORAGE MODULE

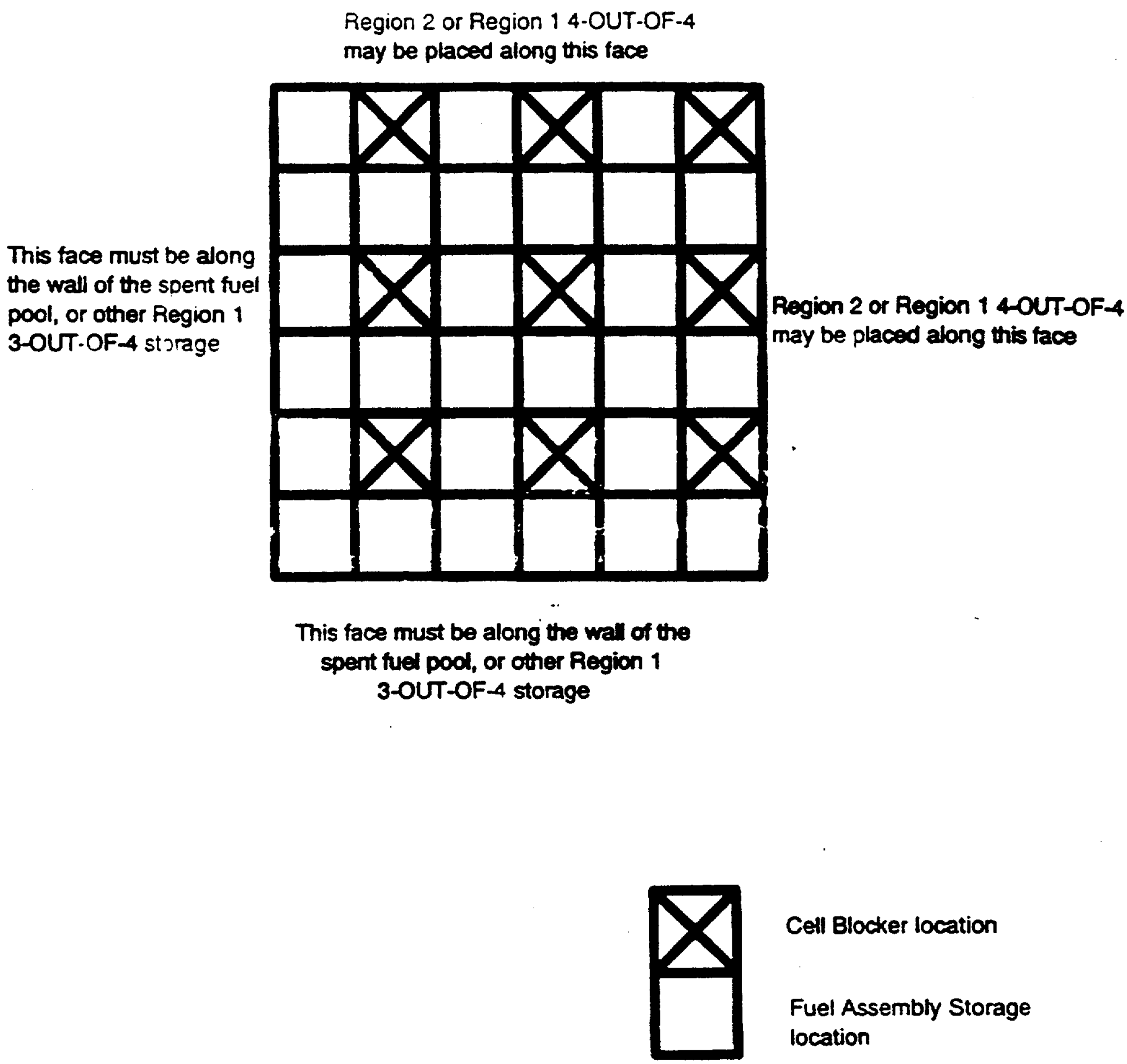

FIGURE 3.9-2

REGION 1 3-OUT-OF-4 STORAGE FUEL ASSEMBLY LOADING SCHEMATIC

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FIGURE 3.9-3 **Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 2 Storage Configuration**

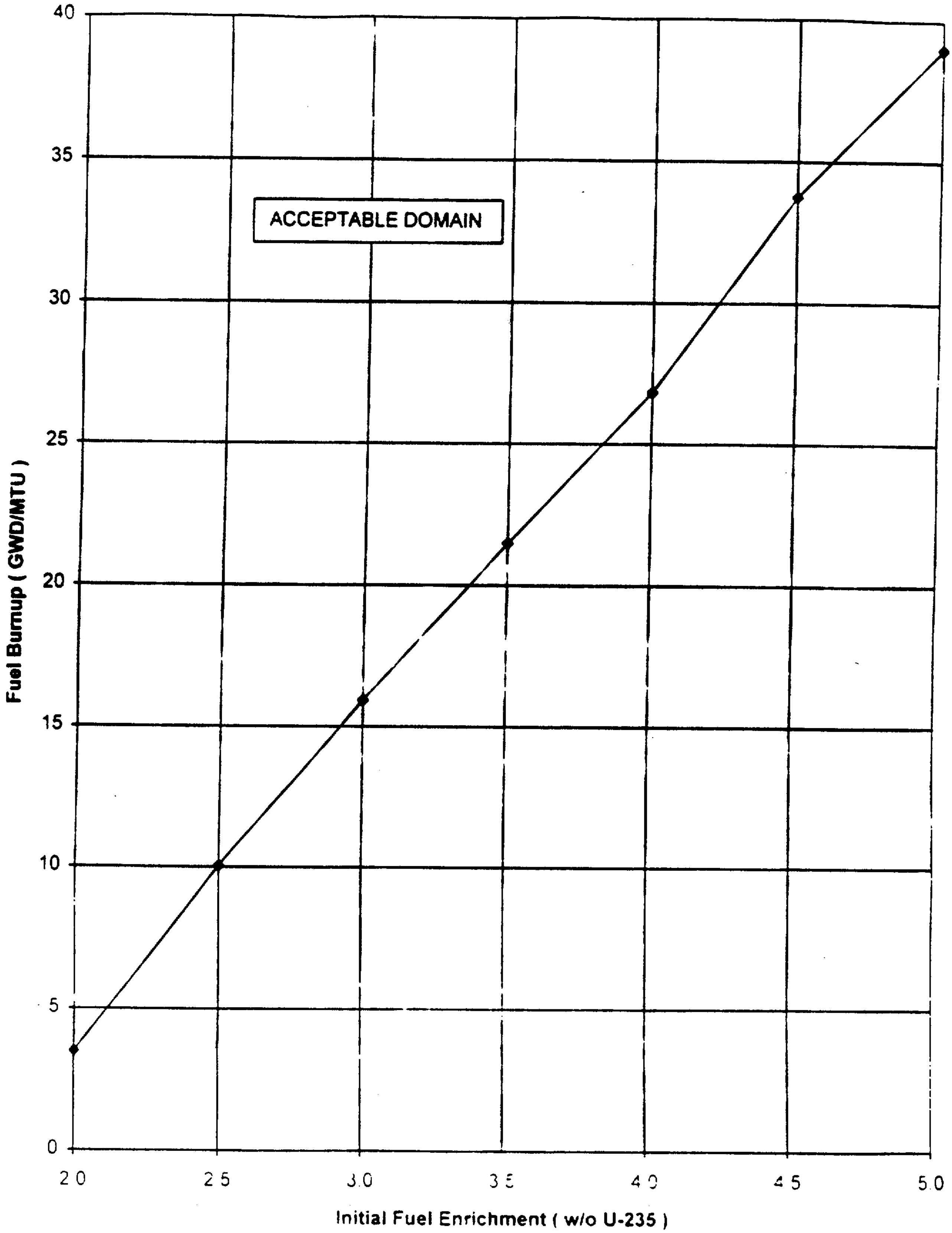
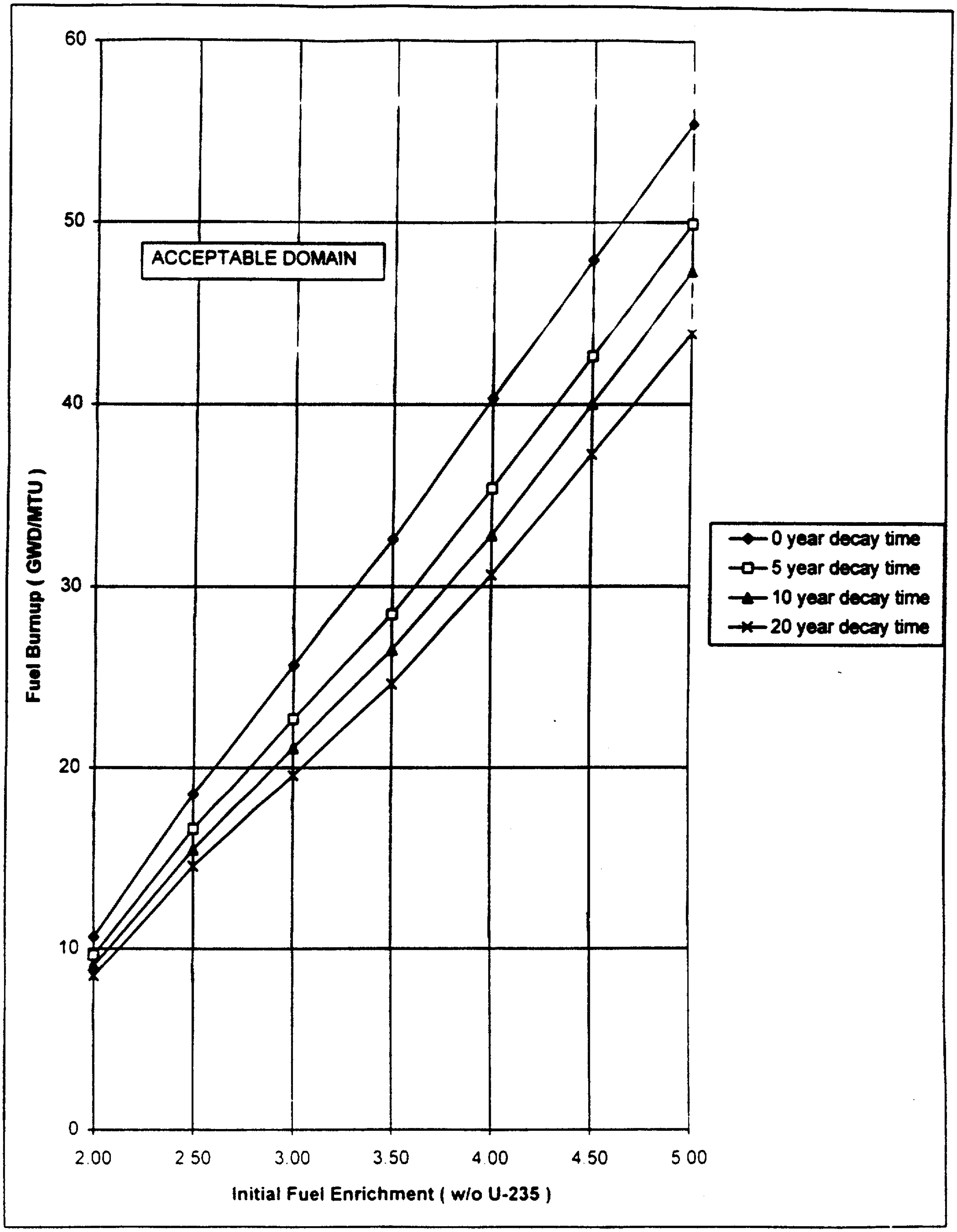


FIGURE 3.9-4 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration



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9.1.1

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2600 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The 2600 ppm provides for boron concentration measurement uncertainty between the spent fuel pool and the RWST. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.1.2 Boron Concentration in Spent Fuel Pool

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During normal Spent Fuel Pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than or equal to 0.95 in an unborated water environment due to the geometry of the rack spacing and the presence of Boraflex neutron absorber in the spent fuel racks. Seismic analysis has shown that there is a possibility that the Boraflex absorber could degrade following a seismic event greater in magnitude than an Operating Basis Earthquake (OBE). At least 1500 ppm boron in Spent Fuel Pool is required in anticipation that a seismic event could cause a loss of Boraflex integrity. If, in addition to a loss of Boraflex, a single misplaced fuel assembly is postulated, then a minimum of 1750 ppm boron is required. The 1750 ppm boron concentration requirement bounds conditions for a loss of all Boraflex in the fuel racks.

The boron requirement in the spent fuel pool also ensures that in the event of a fuel assembly handling accident involving either a dropped or misplaced fuel assembly, the K_{eff} of the spent fuel storage rack will remain less than or equal to 0.95.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

20

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3/4 9.1.2 Boron Concentration in Spent Fuel Pool

During normal spent fuel pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than or equal to 0.95 in an unborated water environment. This is accomplished in Region 1, 2, and 3 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in some fuel storage regions, the limits on fuel burnup, fuel enrichment and minimum fuel decay time, and the use of blocking devices in certain fuel storage locations.

The boron requirement in the spent fuel pool specified in 3.9.1.2 ensures that in the event of a fuel assembly handling accident involving either a single dropped or misplaced fuel assembly, the K_{eff} of the spent fuel storage racks will remain less than or equal to 0.95.

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3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

The limitations on the Fuel Building Exhaust Filter System ensure that all radioactive iodine released from an irradiated fuel assembly and storage pool water will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage. The filtration system removes radioiodine following a fuel handing or heavy load drop accident. Noble gases would not be removed by the system. Other radionuclides would be scrubbed by the storage pool water. Iodine-131 has the longest half-life: ~8 days. After 60 days decay time, there is essentially negligible iodine and filtration is unnecessary.

3/4.9.13 SPENT FUEL POOL - REACTIVITY

The limitations described by Figure 3.9-1 ensure that the reactivity of fuel assemblies introduced into Region II are conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the enrichment and burn-up limits of Figure 3.9-1 have been maintained for the fuel assembly.

During normal Spent Fuel Pool operation, the spent fuel racks are capable of maintaining k_{eff} at less than 0.95 in an unborated water environment due to the geometry of the rack spacing and the presence of Boraflex neutron absorber in the spent fuel racks. Due to radiation induced embrittlement, there is a possibility that the Boraflex absorber could degrade following a seismic event. At least 1500 ppm boron in the Spent Fuel Pool is required in anticipation that a seismic event could cause a complete loss of all Boraflex. If, in addition to a loss of Boraflex, a single misplaced fuel assembly is postulated, then a minimum of 1750 ppm boron is required. The 1750 ppm boron concentration requirement bounds conditions for a loss of all Boraflex in the fuel racks.

The action requirements of this specification recognize the possibility of a seismic event which could degrade the Boraflex neutron absorber in the spent fuel racks. Seismic analysis has shown that there is a possibility that the Boraflex absorber could degrade following a seismic event greater in magnitude than an

Replace w Issue G

3/4.9.13 SPENT FUEL POOL - REACTIVITY (continued)

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Operating Basis Earthquake (OBE). The action statement specifies that following a seismic event at the OBE level or greater, which is approximately one-half the Safe Shutdown Earthquake (SSE) level, action will be taken to determine the condition of the Boraflex. Once a seismic event of greater than or equal to an OBE has occurred, then the boron in the Spent Fuel Pool will be credited to maintain k_{eff} less than or equal to 0.95. The specification requires that dilution paths to the Spent Fuel Pool be closed and administratively controlled until the racks can be inspected and the condition of the Boraflex can be determined. The specification also assumes that piping systems external to the Spent Fuel Pool are mounted such that they remain leak tight following an earthquake up to the level of an SSE, or will not direct water into the Spent Fuel Pool should they leak, or have been isolated from flow to prevent leakage into the Spent Fuel Pool.

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN

1

The limitations of this specification ensure that the reactivity conditions of the Region ~~X~~ 1 3-OUT-OF-4 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 ~~N~~ 1 3-OUT-OF-4 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 assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region ~~X~~ 1 storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the ~~Region I/Region II boundary~~ Region 1 3-OUT-OF-4 Boundary to other Storage Regions and minimize the number of boundaries where a fuel misplacement incident can occur.

INSERT G

3/4 9.13 Spent Fuel Pool Reactivity

During normal spent fuel pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than or equal to 0.95 in an unborated water environment.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 3-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks a maximum nominal 5 weight percent fuel enrichment, and the use of blocking devices in certain fuel storage locations, as specified by the interface requirements shown in Figure 3.9-2.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 4-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-1.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 2 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-3.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 3 storage racks by the combination of geometry of the rack spacing, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-4. Fixed neutron absorbers are not credited in the Region 3 fuel storage racks.

The limitations described by Figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 ensure that the reactivity of the fuel assemblies stored in the spent fuel pool are conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the fuel enrichment, fuel burnup, fuel decay times, and fuel interface restrictions specified in Figures 3.9-1, 3.9.2, 3.9-3 and 3.9-4 are complied with.

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5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water

b. A nominal 10.35-inch center-to-center distance between fuel assemblies placed in the storage racks.

c. Fuel assemblies stored in Region I of the spent fuel pool may have a maximum nominal fuel enrichment of up to 5. weight percent U_{235}. Region I is designed to ... fuel in a 3-out-of-4 array with the 4th storage location blocked as shown in Figure 3.9-2.

d. Fuel assemblies stored in Region II of the spent fuel pool may have a maximum nominal fuel enrichment of up to 5.0 weight percent, conditional upon compliance with Figure 3.9-1 to ensure that the design burnup of the fuel 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 45 feet.

CAPACITY

5.6.3 ~~The spent fuel storage pool contains 756 storage locations of which a maximum of 100 locations will be blocked~~.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

See Insert H

25

INSERT H

5.6.1.1 The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a K_{eff} less than or equal to 0.95 when flooded with unborated water. The 3 storage rack Regions are:

a. Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and can store fuel in 2 storage configurations:
(1) With credit for fuel burnup as shown in Figure 3.9-1, fuel may be stored in a "4-OUT-OF-4" storage configuration.
(2) With credit for every 4th location blocked and empty of fuel, fuel up to 5 weight percent nominal enrichment, regardless of fuel burnup, may be stored in a "3-OUT-OF-4" storage configuration. Fuel storage in this configuration is subject to the interface restrictions specified in Figure 3.9-2.

b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and with credit for fuel burnup as shown in Figure 3.9-3, fuel may be stored in all available Region 2 storage locations.

c. Region 3, a nominal 10.35 inch center to center distance, with credit for fuel burnup and fuel decay time as shown in Figure 3.9-4, fuel may be stored in all available Region 3 storage locations. The Boraflex contained inside these storage racks is not credited.

5.6.3 The spent fuel storage pool contains 350 Region 1 storage locations, 673 Region 2 storage locations and 756 Region 3 storage locations, for a total of 1779 total available fuel storage locations. An additional Region 2 rack with 81 storage locations may be placed in the spent fuel pool, if needed. With this additional rack installed, the Region 2 storage capacity is 754 storage locations, for a total of 1860 total available fuel storage locations

(End of TS Change)

Docket No. 50-423
B17343

Attachment 2

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specification
Spent Fuel Pool Rer..ck (TSCR 3-22-98)

Retypeo Pages

March 1999

RETYPE OF PROPOSED REVISION

Refer to the attached retype of the proposed revision to the Technical Specifications. The attached retype reflects the currently issued version of the Technical Specifications. Pending Technical Specification revisions or Technical Specification revisions issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

VENTING

1.39 VENTING shall be 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.

SPENT FUEL POOL STORAGE PATTERNS:

STORAGE PATTERN

1.40 STORAGE PATTERN refers to the blocked location in a Region 1 fuel storage rack and all adjacent and diagonal Region 1 (or Region 2) cell locations surrounding the blocked location. The blocked location is for criticality control.

3-OUT-OF-4 and 4-OUT-OF-4

1.41 Region 1 spent fuel racks can store fuel in either of 2 ways:

(a) Areas of the Region 1 spent fuel racks with fuel allowed in every storage location are referred to as the 4-OUT-OF-4 Region 1 storage area.

(b) Areas of the Region 1 spent fuel racks which contain a cell blocking device in every 4th location for criticality control, are referred to as the 3-OUT-OF-4 Region 1 storage area. A STORAGE PATTERN is a subset of the 3-OUT-OF-4 Region 1 storage area.

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

ALLOWED POWER LEVEL

1.43 APL^{ND} is the minimum allowable nuclear design power level for base load operation and is specified in the COLR.

1.44 APL^{BL} is the maximum allowable power level when transitioning into base load operation.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be maintained uniform, and greater than or equal to 800 ppm.

Applicability

During all fuel assembly movements within the spent fuel pool.

Action

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

SURVEILLANCE REQUIREMENTS

4.9.1.2 Verify that the soluble boron concentration is greater than or equal to 800 ppm prior to any movement of a fuel assembly into or within the spent fuel pool, and every 7 days thereafter during fuel movement.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2200 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2200 pounds over the fuel storage pool shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation. Administrative controls may be used in lieu of crane interlocks and physical stops for handling fuel racks, spent fuel pool gates, or loads less than 2200 pounds.

3/4.9.13 SPENT FUEL POOL - REACTIVITY

LIMITING CONDITION FOR OPERATION

3.9.13 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 assemblies are in the spent fuel pool.

ACTION: With k_{eff} greater than 0.95:

a. Borate the Spent Fuel Pool until k_{eff} is less than or equal to 0.95, and

b. Initiate immediate action to move any fuel assembly which does not meet the requirements of Figures 3.9-1, 3.9-3 or 3.9-4, to a location for which that fuel assembly is allowed.

SURVEILLANCE REQUIREMENTS

4.9.13.1.1. Ensure that all fuel assemblies to be placed in Region 1 "4-OUT-OF-4" fuel storage are within the enrichment and burnup limits of Figure 3.9-1 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1.2. Ensure that all fuel assemblies to be placed in Region 2 fuel storage are within the enrichment and burnup limits of Figure 3.9-3 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1.3. Ensure that all fuel assemblies to be placed in Region 3 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-4 by checking the fuel asssembly's design, decay time, and burn-up documentation.

REFUELING OPERATIONS

SPENT FUEL POOL - STORAGE PATTERN

LIMITING CONDITION FOR OPERATION

3.9.14 Each STORAGE PATTERN of the Region 1 spent fuel pool racks shall require that:

a. Prior to storing fuel assemblies in the STORAGE PATTERN per Figure 3.9-2, the cell blocking device for the cell location must be installed.

b. Prior to removal of a cell blocking device from the cell location per Figure 3.9-2, the STORAGE PATTERN must be vacant of all stored fuel assemblies

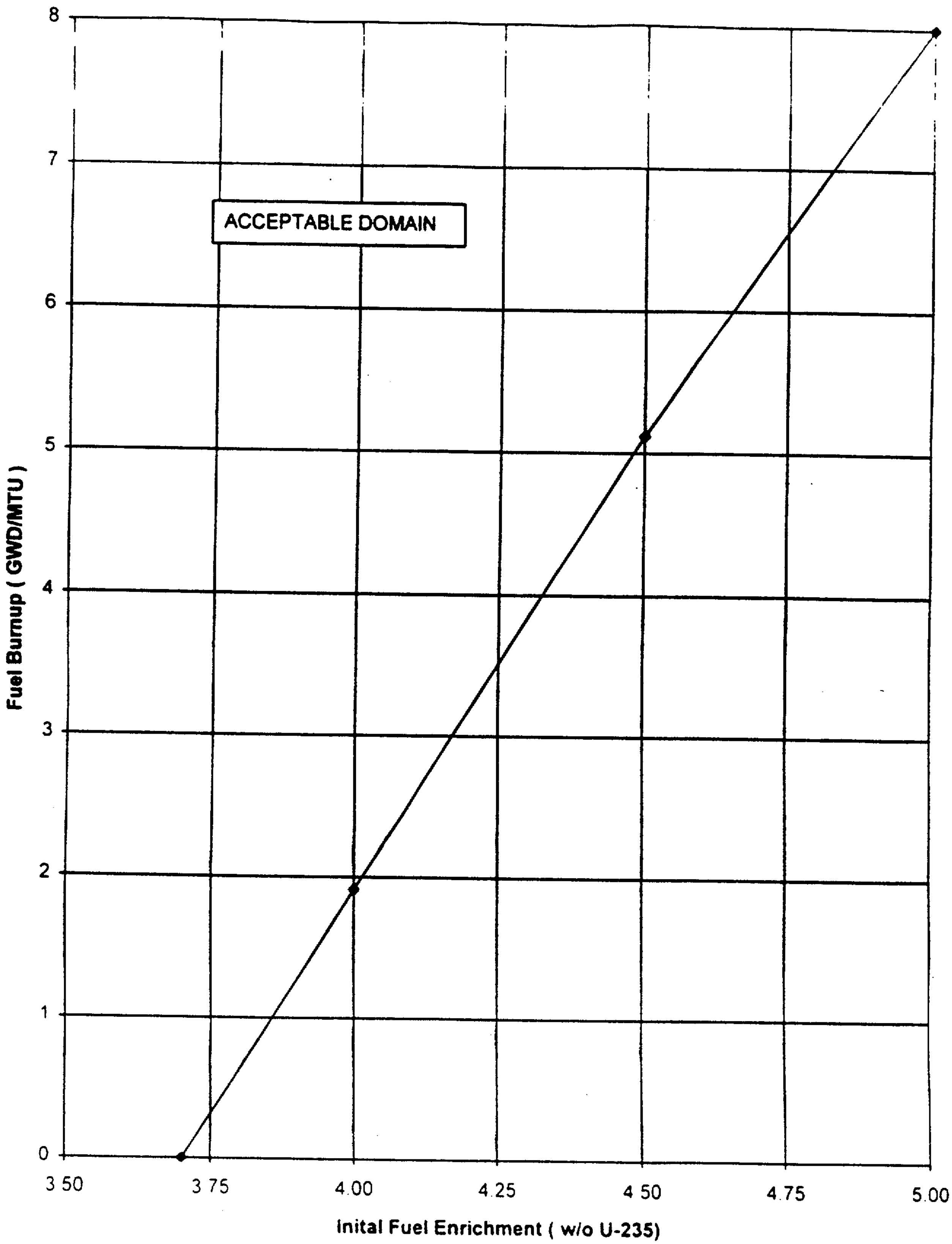
APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

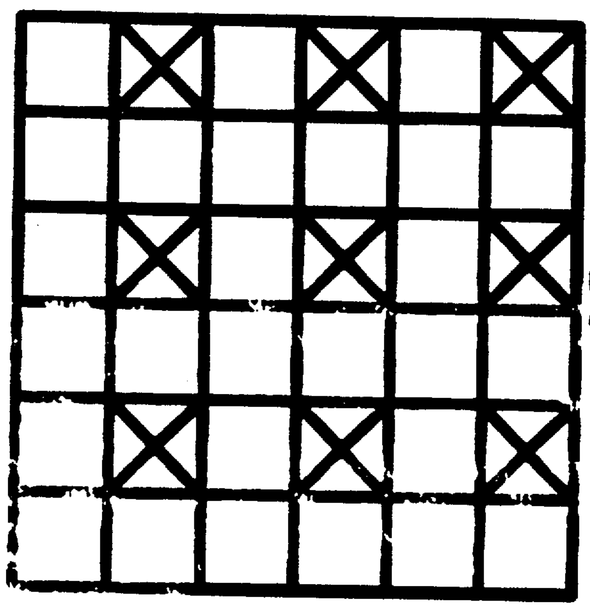
ACTION: Take immediate action to comply with 3.9.14(a), (b).

SURVEILLANCE REQUIREMENTS

4.9.14 Verify that 3.9.14 is satisfied with no fuel assemblies stored in the STORAGE PATTERN prior to installing and removing a cell blocking device in the spent fuel racks.

FIGURE 3.9-1 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 1 4-OUT-OF-4 Fuel Storage Configuration





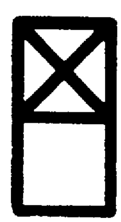


FIGURE 3.9-2

REGION 1 3-OUT-OF-4 STORAGE FUEL ASSEMBLY LOADING SCHEMATIC

FIGURE 3.9-3 Minimum Fuel Assembly Burnup Versus Nominal Initial Enrichment for Region 2 Storage Configuration

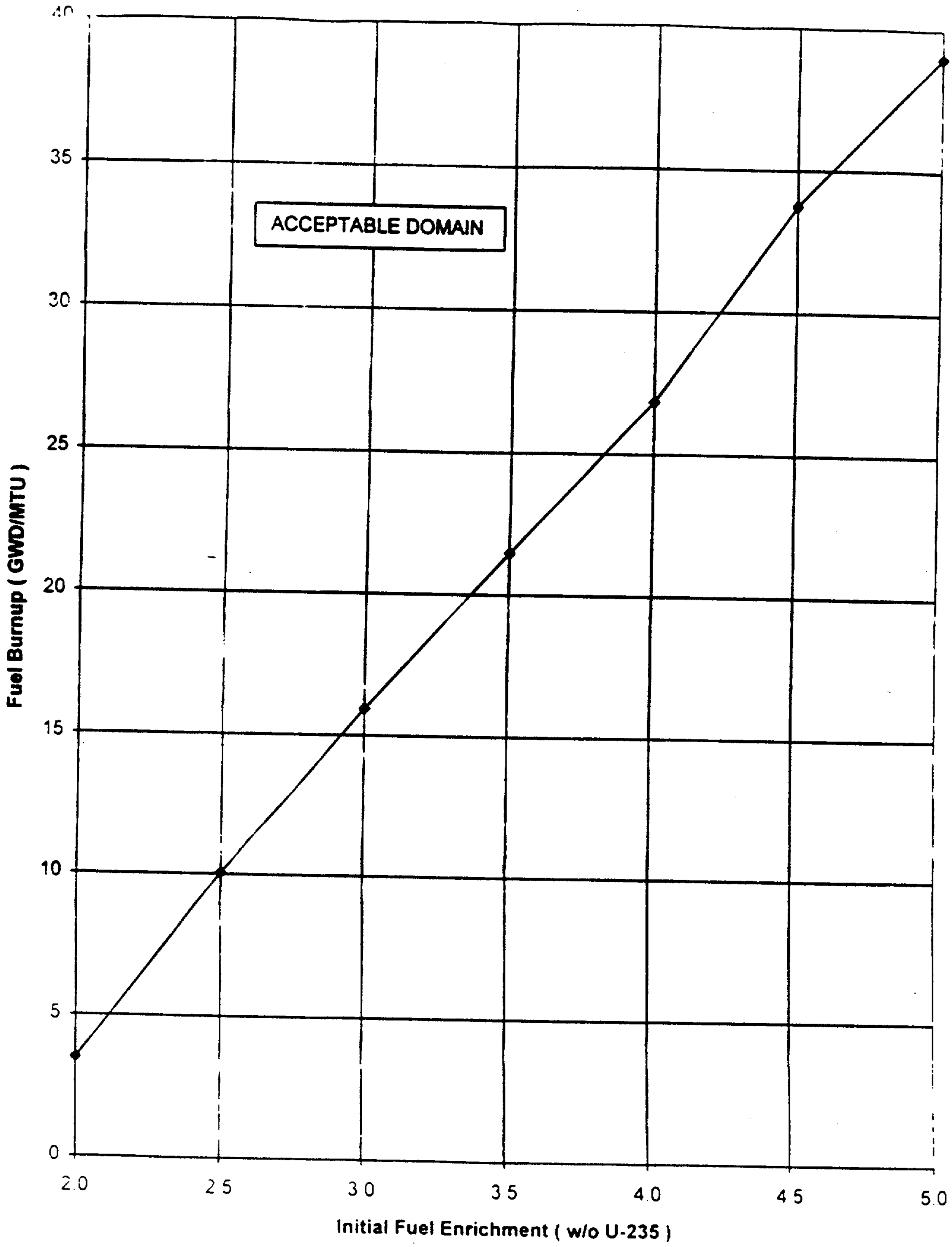
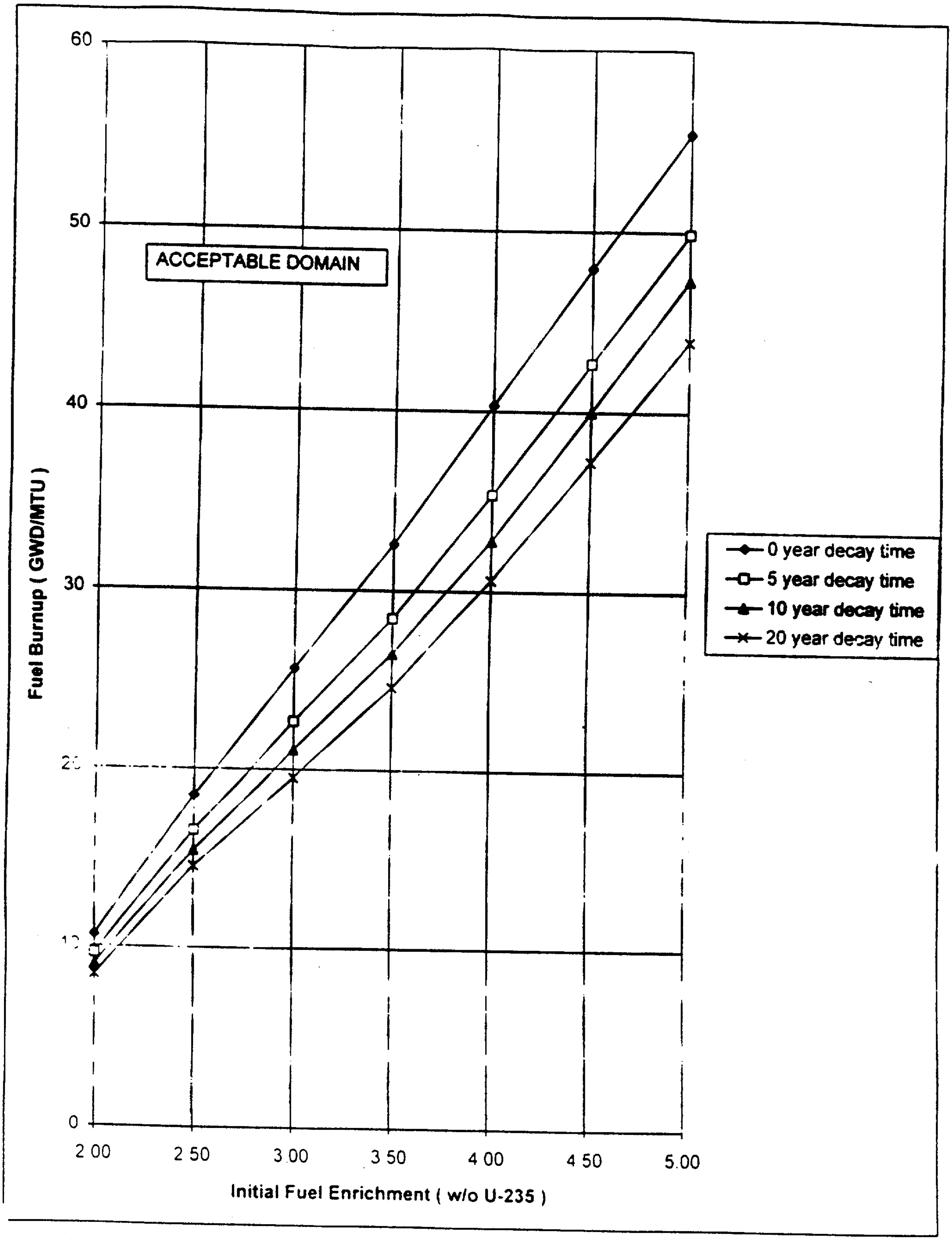


FIGURE 3.9-4 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration



3/4.9.1.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2600 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The 2600 ppm provides for boron concentration measurement uncertainty between the spent fuel pool and the RWST. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.1.2 Boron Concentration in Spent Fuel Pool

During normal spent fuel pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than or equal to 0.95 in an unborated water environment. This is accomplished in Region 1, 2, and 3 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in some fuel storage regions, the limits on fuel burnup, fuel enrichment and minimum fuel decay time, and the use of blocking devices in certain fuel storage locations.

The boron requirement in the spent fuel pool specified in 3.9.1.2 ensures that in the event of a fuel assembly handling accident involving either a single dropped or misplaced fuel assembly, the K_{eff} of the spent fuel storage racks will remain less than or equal to 0.95.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

The limitations on the Fuel Building Exhaust Filter System ensure that all radioactive iodine released from an irradiated fuel assembly and storage pool water will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage. The filtration system removes radioiodine following a fuel handing or heavy load drop accident. Noble gases would not be removed by the system. Other radionuclides would be scrubbed by the storage pool water. Iodine-131 has the longest half-life: ~8 days. After 60 days decay time, there is essentially negligible iodine and filtration is unnecessary.

3/4.9.13 SPENT FUEL POOL - REACTIVITY

During normal spent fuel pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than or equal to 0.95 in an unborated water environment.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 3-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, a maximum nominal 5 weight percent fuel enrichment, and the use of blocking devices in certain fuel storage locations, as specified by the interface requirements shown in Figure 3.9-2.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 4-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-1.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 2 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-3.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 3 storage racks by the combination of geometry of the rack spacing, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-4. Fixed neutron absorbers are not credited in the Region 3 fuel storage racks.

3/4.9.13 SPENT FUEL POOL - REACTIVITY (continued)

The limitations described by Figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 ensure that the reactivity of the fuel assemblies stored in the spent fuel pool are conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the fuel enrichment, fuel burnup, fuel decay times, and fuel interface restrictions specified in Figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 are complied with.

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity conditions of the Region 1 3-OUT-OF-4 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 1 3-OUT-OF-4 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 assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region 1 storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the Region 1 3-OUT-OF-4 Boundary to other Storage Regions and minimize the number of boundaries where a fuel misplacement incident can occur.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a K_{eff} less than or equal to 0.95 when flooded with unborated water. The storage rack Regions are:

a. Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and can store fuel in 2 storage configurations:

(1) With credit for fuel burnup as shown in Figure 3.9-1, fuel may be stored in a "4-OUT-OF-4" storage configuration.

(2) With credit for every 4th location blocked and empty of fuel, fuel up to 5 weight percent nominal enrichment, regardless of fuel burnup, may be stored in a "3-OUT-OF-4" storage configuration. Fuel storage in this configuration is subject to the interface restrictions specified in Figure 3.9-2.

b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and with credit for fuel burnup as shown in Figure 3.9-3, fuel may be stored in all available Region 2 storage locations.

c. Region 3, a nominal 10.35 inch center to center distance, with credit for fuel burnup and fuel decay time as shown in Figure 3.9-4, fuel may be stored in all available Region 3 storage locations. The Boraflex contained inside these storage racks is not credited.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

CAPACITY

5.6.3 The spent fuel storage pool contains 350 Region 1 storage locations, 673 Region 2 storage locations and 756 Region 3 storage locations, for a total of 1779 total available fuel storage locations. An additional Region 2 rack with 81 storage locations may be placed in the spent fuel pool, if needed. With this additional rack installed, the Region 2 storage capacity is 754 storage locations, for a total of 1860 total available fuel storage locations.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

Docket No. 50-423
B17343

Attachment 5

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specification
Spent Fuel Pool Rerack (TSCR 3-22-98)

Background and Safety Summary

March 1999

Background

Millstone Unit No. 3 received its low power operating licensing in November, 1985. The plant began operations with spent fuel pool racks in their present configuration, which is 21 free standing spent fuel racks with a total storage capacity of 756 fuel assemblies. These racks use the silicone polymer Boraflex as the neutron absorption material.

At present, NNECO is contracted to the U. S. Department of Energy (DOE) to take Millstone Unit No. 3 spent fuel. However, the DOE has not yet begun taking spent fuel from reactor sites. When the DOE begins accepting spent fuel, they plan to accept the oldest spent fuel first. Because Millstone Unit No. 3 was licensed relatively recently, it will be among the last reactor sites to begin its spent fuel shipments to the DOE. Because Millstone Unit No. 3 will lose full core reserve capability in about two years, the plant must increase onsite fuel storage capacity.

NNECO has evaluated spent fuel storage alternatives that have been licensed by the NRC and could be feasible for use at Millstone Unit No. 3. The result of the evaluation is that reracking the Millstone Unit No. 3 spent fuel pool is currently the most cost effective alternative. This increase in spent fuel storage capacity would preserve full core reserve discharge capability approaching the end of its current operating license in the year 2025.

Summary

Millstone Unit No. 3 must rerack its spent fuel pool to maintain full core reserve capability. NNECO proposes to achieve this goal by installing two types of additional higher density spent fuel racks into the spent fuel pool. Existing spent fuel racks will remain in the pool, but are reanalyzed to only accept fuel lower in reactivity than they are licensed to accept at present. The proposed additional racks will have a closer assembly to assembly spacing to help maximize fuel storage capacity.

The planned spent fuel pool storage expansion involves licensing 15 new rack modules for insertion into the Millstone Unit No. 3 spent fuel pool. The expansion will leave in place all of the existing 21 spent fuel racks that are in the Millstone Unit No. 3 spent fuel pool. After the expansion, the pool will contain three distinct administratively controlled storage regions as shown in attached Figure 1. Each region is characterized by a nominal center-to-center spacing of the cells. The new cells will contain a fixed neutron absorber for primary reactivity control. The new racks will be grouped in Regions 1 and 2. The existing racks that will remain in place will be designated as Region 3.

Region 1 and Region 2 racks will contain Boral as the neutron absorbing material. The Boral absorbers are to be sized to fully shadow the assembly total active fuel length.

The existing Region 3 racks contain the silicon rubber polymer, Boraflex, as the neutron absorbing material. But no credit is taken for Boraflex in the criticality analysis for Region 3.

Region 1 racks have the capacity to store up to 350 fuel assemblies. Region 1 can store assemblies with a nominal 5.0 w/o U-235 enrichment in a 3-out-of-4 configuration without restriction on burnup. The 3-out-of-4 configuration utilizes a fuel cell blocker for critically control. Region 1 can also store assemblies in an 4-out-of-4 storage configuration with burnup/enrichment restrictions. Region 1 is sized to accommodate an emergency core offload

Region 2 racks will be licensed to store 754 assemblies. The storage in Region 2 racks will have more restrictive burnup/enrichment restrictions than Region 1 racks and use a 4-out-of-4 storage configuration.

Region 3 racks can store 756 assemblies. The storage in Region 3 racks will have more restrictive burnup/enrichment restrictions than Region 2 racks. Region 3 racks will allow credit for decay of fissile plutonium and buildup of americium, which reduce reactivity, as a function of decay time. Other domestic nuclear plants have been licensed for decay time credit.

The proposed Millstone Unit No. 3 rerack project will increase the licensed storage capacity from 756 to 1,860 fuel assemblies, which will provide sufficient licensed capacity to allow operation approaching the end of the current plant operating license in the year 2025. As shown in Figure 1, Millstone Unit No. 3 does not plan to install the southern most Region 2 rack at this time; it will be installed if and when necessary. The structural analyses, seismic analyses, rerack analyses and the Significant Hazards Consideration assume that this rack is installed, which bounds the pool configuration of the rack not being installed.

All rack modules in the Millstone Unit No. 3 pool will be free-standing and self-supporting. This includes the existing racks that will comprise Region 3 after the transition phase. After installation, rack locations will be surveyed to ensure proper positioning. Attachments 5 and 6 detail the proposed rack configuration in the reracked pool.

With the expanded capacity, the spent fuel pool cooling system will be required to remove an increased heat load while maintaining the pool water temperature within the design limit. The maximum heat load typically develops from the residual heat in the pool after the last core offload at the end of plant life. NNECO has reanalyzed spent fuel pool thermal performance. The fuel pool thermal performance analysis, as it applies to bulk pool temperature and equipment under higher heat loads, is under a separate NNECO letter dated January 18, 1999 (B17004). However, this proposed amendment request does analyze local temperature peaks.

Seismic and structural analyses were performed for the racks and pool structure. The racks and pool structure will maintain their function and ensure the integrity, subcriticality margin, and coolability of fuel assemblies under postulated seismic events and mechanical accidents.

The following addresses the safety issues arising from the reracking and proposed revisions to the Technical Specifications of Millstone Unit No. 3. The scope of the technical analysis supporting this evaluation focused mainly on the final licensed configuration of the expanded spent fuel pool storage space, including all Region 2 racks.

Mechanical Design Evaluation

The new fuel rack design has been evaluated with respect to the mechanical and material qualifications, neutron poison and poison surveillance requirements, fuel handling qualifications, fuel interfaces, and accident considerations.

The proposed additional spent fuel racks are free standing and self supporting. The principal construction materials are ASME SA240-304L for stainless steel sheet and plate stock, and internally threaded support legs. The externally threaded support spindle is SA564-630 precipitation hardened stainless steel (heat treated to 1,100°F). The only non-stainless steel material in the racks is the Boral which is a composite of boron carbide and type 1100 alloy aluminum, within a layer of type 1100 aluminum. The governing quality assurance requirements for fabrication of the racks meet the quality assurance and quality control of 10CFR50, Appendix B requirements.

For primary nuclear criticality control in the new racks, the racks will integrate a fixed neutron absorber into its structure. The absorber, trade name Boral, is a boron carbide and aluminum-composite sandwich. It is chemically inert and has a long history of applications in the spent fuel pool environments where it has maintained its neutron attenuation capability under thermal loads. Boral is manufactured under the control of a quality assurance program which conforms to the requirements of 10CFR50, Appendix B. Region 3 racks contain Boraflex as the fixed neutron absorber. However, Boraflex will no longer be credited per this request.

The support legs on the racks will allow for remote leveling and alignment of the rack modules to accommodate variations in the floor flatness. A thick bearing pad will be interposed between the rack pedestals and the floor to distribute the dead load over a wider support area.

The rack structural performance with respect to the impact and seismic loads, as well as the subcritical configuration, has been analyzed. The analysis included an accidental drop of a fuel assembly during movement to a storage location, and induced tensile loads on the rack arising from a stuck assembly in the storage cell. It has been shown that these accidents will not invalidate the mechanical design and material selection criteria to safely store spent fuel in a coolable and subcritical configuration in any region. The fuel will maintain its structural integrity and remain subcritical.

Testing procedures will be developed to periodically verify acceptable performance of the Boral. The testing will use Boral coupons to verify the quality and presence of a sufficient amount of neutron absorber in the racks to assure subcriticality margin. The testing will not extend to the Boraflex absorber in Region 3 since the Boraflex is not credited in the criticality analysis.

Criticality Considerations

The proposed additional spent fuel racks are designed to maintain the required subcriticality margin when fully loaded with fuel of the maximum permissible reactivity for a given storage region, and in unborated water at a temperature within the normal operating range corresponding to the highest reactivity. For reactivity control in Region 1 and 2 racks, Boral panels will be used. The panels are sized to fully shadow the active fuel height of all assembly designs stored in the pool. The panels will be held in place and protected against damage by a stainless steel jacket that is welded to the cell walls. In Region 1, the panels will be mounted on the outside faces of each cell. In Region 2, the panels will be mounted either on the exterior or on the interior of the cells, in an alternating pattern. The existing racks, in what will become Region 3, contain Boraflex as the neutron absorber. However, no credit is taken for Boraflex in the criticality analysis.

The storage of spent fuel in each region will be controlled by the criteria defining the maximum permissible reactivity. Region 1 can store fuel assemblies of up to 5.0 w/o nominal enrichment, regardless of burnup, in a 3-out-of-4 storage array subject to a blocking/interface restriction. Region 1 can store fuel in a 4-out-of-4 array subject to proposed burnup/enrichment limits.

Region 2 can store fuel in a 4-out-of-4 array subject to proposed burnup/enrichment limits which are more restrictive than those in Region 1.

Region 3 can store fuel in a 4-out-of-4 array subject to the burnup/enrichment/decay time limits. Region 3 has the most restrictive burnup/enrichment limits of the 3 regions. Also, Region 3 burnup limits decrease with increased fuel decay time.

If a fuel assembly does not meet the requirements for storage in either Region 2 or 3, then it must be stored in Region 1.

The USNRC guidelines and the ANSI standards specify that the margin of safety for criticality be determined by the maximum neutron multiplication factor k_{eff} less than or equal to 0.95, including uncertainties, for all normal and accident conditions. The analysis has shown that this criterion is always maintained under all postulated accidents. The accidents and malfunctions evaluated included a dropped fuel assembly onto fuel racks, impact on criticality of water temperature and density effects, impact on criticality of eccentric positioning of a fuel assembly within the rack, and misloading of the most reactive assembly in a Region 1, Region 2. or Region 3 rack (highest reactivity error).

The proposed Technical Specifications will require a minimum concentration of 800 ppm of soluble boron in the pool water during fuel movement to assure k_{eff} will remain less than or equal to 0.95 assuming a dropped or misloaded fuel assembly. The surveillance interval for this soluble boron concentration in the proposed Technical Specifications is consistent with Westinghouse improved STS 3.7.16.

For spent fuel pool water temperature effects, the most reactive spent fuel pool water temperature in the normal operating range was used in the criticality calculations. The criticality analysis uses a range of 32°F to 160°F to bound the fuel pool normal operating water temperature span. For Regions 1 and 2, fuel pool water temperatures in excess of 160°F are less reactive. For Region 3, the most reactive temperature is boiling. However, fuel pool water temperatures in excess of 160°F are outside of the design basis of the fuel pool cooling system. The fuel pool cooling system is capable of maintaining the fuel pool temperature less than 160°F.

Thermal Hydraulics and Pool Cooling

A comprehensive thermal-hydraulic evaluation of the expanded spent fuel pool has been done to analyze its thermal performance to support a separate licensing amendment request dated January 18, 1999 (B17004). This comprehensive analysis supports treating full core offloads as a normal evolution. The submittal's assumed heat load bounds the heat load associated with this rerack licensing amendment request and NRC approval of the January 18, 1999, submittal is required prior to approval of this rerack licensing amendment. However, this rerack licensing amendment request calculated the local peak water temperature and local peak clad temperature which is based on the January 18, 1999 (B17004), submittal heat load.

The peak local water and fuel clad temperatures were computed for the rerack license amendment for the partially blocked hottest channel. The peak local water temperature was well below the boiling temperature at the top of fuel with fuel pool water level at its low level alarm. This analysis assures that flow will remain subcooled which minimizes the potential for fuel damage. Also, the peak clad temperature is well below the temperature where clad damage or a zirconium-water reaction would occur.

Seismic and Structural Evaluation

NNECO has re-evaluated the mechanical and civil structures to address the structural issues resulting from the Millstone Unit No. 3 rerack. The analysis considered the loads from seismic, thermal, and mechanical forces to determine the margin of safety in the structural integrity of the fuel racks, the spent fuel pool, and the pool liner. The loads, load combinations, and acceptance criteria were based on ASME Boiler and Pressure Vessel Code, 1995 Edition, Section III, Subsection NF and NUREG-800, Standard Review Plan (SRP) Section 3.8.4.

a. The storage rack evaluation

The final configuration of the pool will consist of free standing fuel rack modules in all three regions. The seismic analysis has separately evaluated a single free-standing rack as well as the whole pool multi-rack structure in 3-dimensions. The analyses were based on the simulation of the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE) in accordance with SRP 3.7.1 requirements.

The following computed stress loadings were compared against the allowable stress loadings in ASME Boiler and Pressure Vessel Code, 1995 Edition, Section III, Subsection NF:

- Maximum Fuel Storage Cell Region Stress Factor - The maximum stress factor for every rack was computed to be within allowable limits.

- Maximum Pedestal Thread Shear Stress - The maximum pedestal thread shear stress was computed to be within allowable limits.

- Impact Load Between Fuel Assembly and Fuel Storage Wall - The assembly is postulated to rattle against the cell wall during an SSE creating a load between the assembly and wall. The maximum load on the cell wall was computed to be well within allowable limits.

  Impacts Between Adjacent Racks - The analysis shows that rack movement during the postulated SSE will not lead to impacts between rack cell walls of proposed additional racks, and between proposed additional racks and existing racks. The analysis only predicts rack-to-rack impacts between proposed additional racks at the 3/4 inch baseplate which extends out of the bottom of the these racks. The highest computed impact stress would cause very little or no deformation of the baseplates. Rack storage cells, fuel, and Boral would be undamaged.

- Baseplate to Fuel Rack Storage Cell Weld Stresses - The maximum stress on a weld between a base plate and a fuel storage cell is computed to be within allowable limits.

- Baseplate to Fuel Rack Pedestal Weld Stresses - The maximum stress on a weld between a base plate and a pedestal is computed to be within allowable limits.

- Fuel Rack Storage Cell to Fuel Rack Storage Cell Weld Stresses - The maximum stress on a weld between fuel storage cells is computed to be within allowable limits.

- Rack Fatigue - The cumulative damage factor due to rack stress fatigue is computed to be within allowable limits.

The analyses results show that each of the above factors are within their allowable limits. Thus, it is concluded that the racks will maintain their integrity, protect the fuel and Boral from damage, and maintain subcriticality margin and coolability under all postulated design conditions.

b. Pool structural evaluation

The pool structure has been analyzed using a 3-D finite element model seismically accelerated with a synthetic time history motion applied just below the base mat level. The analyses used the individual dead, live, thermal, and seismic loads and load combinations required by NUREG-800, SRP Section 3.8.4. The analyses show that the pool structure satisfies these required load combinations, and will maintain its integrity and protect racks and fuel for all postulated scenarios.

The following loadings were compared against allowable loadings:

- Pool Walls - The analysis computed the limiting safety margin for the fuel pool for both bending strength and shear strength on the four fuel pool walls, the transfer canal wall, and the cask pit north and west walls. The smallest limiting safety margin for both bending strength and shear strength occurred on the cask pit west wall, and were well within allowable limits. All other computed safety margins were greater. Thus, it is concluded that the structural capacity of the fuel pool is maintained under all required load combinations.

- Base Slab - This massive structural slab supporting the pool structure, is heavily reinforced, continuous throughout the Fuel Building area of concern, and supports the whole building. The load additions to the base slab due to the rerack are primarily compressive loads that are supported on bedrock grade. These load increases are very small in comparison to the base slab capacity. Therefore, a simplifying assumption is that the base mat remains adequate in total. Local stresses on the basemat from fuel rack bearing pads due to mechanical accidents and seismic loadings are discussed subsequently.

- Pool Liner - The pool liner will maintain its integrity during a postulated seismic event. During the postulated seismic occurrence, the fuel rack pedestals will impart loads onto the pool floor. The analysis found that these loads will not tear or cause fatigue failure of the fuel pool's stainless steel liner and welds.

- Bearing Pads - Bearing pad pressure on the fuel pool slab meets the required limits after a postulated seismic occurrence and for all loading conditions. Bearing pads are placed between the pedestal base and fuel pool liner to protect the liner from high localized dynamic loadings, and to distribute the load imparted to the slab. During a seismic event, fuel rack pedestals impact the bearing pads transferring pedestal loads to the liner. Bearing pad dimensions are set to assure that the average pressure on the pool slab surface due to static and dynamic loads does not exceed allowable limits on bearing pressures. Two stress factors were computed, the average pressure at the slab/liner interface, and the maximum bending stress at the bearing pad. Both of these stress factors were found within allowable limits. Therefore, the bearing pad design is adequate for all design basis loadings.

c. Mechanical Accidents

In addition to the seismic loads, the racks and the pool liner were also analyzed for mechanical loads under accident conditions. The following accident scenarios were analyzed:

- Fuel Assembly with Control Rod and Handling Tool Drop Onto Racks - The analysis shows that if a fuel assembly with control rod and handling tool drop from above the maximum lift height of the spent fuel bridge hoist onto a rack, only the upper region of the impacted storage cell is damaged, thus protecting the Boral and stored fuel assembly from damage. Also, local thermal hydraulic requirements continue to be met since only minor distortion of the fuel cell geometry will occur.

- Fuel Assembly with Control Rod and Handling Tool Drop Through an Empty Rack Storage Cell Over a Pedestal Location - This scenario assumed the spent fuel bridge hoist drops a fuel assembly with control rod and handling tool from above the maximum lift height into an empty fuel storage cell over a pedestal location. This scenario maximizes the load imparted to the pool liner. The analysis concluded that this scenario would cause negligible rack baseplate deformation and insignificant plastic strain in the liner. Thus, the liner would maintain its integrity.

- Fuel Assembly with Control Rod and Handling Tool Drop Through an Empty Interior Rack Storage Cell - This scenario assumed the spent fuel bridge hoist drops a fuel assembly with control rod and handling tool from above the maximum lift height into an empty interior fuel storage cell. The fuel assembly falls unimpeded through the storage cell until it strikes the rack baseplate at the bottom of the storage cell. This impact is postulated to occur at an interior storage cell location to maximize the predicted baseplate deformation, and produces localized severing of the baseplate/storage cell welds. However, the baseplate still maintains its integrity and prevents the fuel assembly from impacting the liner. Thus, no liner damage occurs.

- Spent Fuel Rack is Dropped onto Fuel Pool Liner During Installation - The analysis concludes that if a rack drops 40 feet onto the liner during installation, liner puncture and small indentations in the pool floor concrete surface would occur. A small rate of water seepage, which is well within makeup capability, could occur. Such seepage is considered minor and procedures exist that direct the operators to initiate emergency make-up to the pool, if necessary. There will also be a contingency procedure to repair liner damage, should it occur, during the rack installation.

- Fuel Assembly Becomes Stuck When Being Removed from Fuel Storage Rack - The analysis shows that the rack structural integrity will not be compromised if a fuel assembly becomes stuck during removal from a rack.

- Fuel Pool Gate Drops onto a Fuel Storage Rack - The transfer canal fuel pool gate will now be moved over fuel racks because Region 1 racks will be installed within several inches of the fuel pool west wall. The cask pit storage gate when being moved does go over existing fuel racks. Both the proposed additional racks and the existing racks were analyzed for a fuel pool gate drop. This analysis demonstrates that a gate drop would not damage a stored fuel assembly (provided the fuel assembly does not contain a control rod assembly or other insert) or cause damage to the neutron absorber material or to the pool liner. In addition, although the upper portion of the impacted rack suffers local deformation, the overall structural integrity of the rack is not compromised; thus the storage array configuration is maintained, and there are no resulting criticality concerns. Nevertheless, the requirements of Technical Specification 3.9.7 will continue to prohibit fuel pool gate movement over fuel assemblies since a gate weighs more than the imposed 2,200 lb. load limit.

- Cask Drop - The consequences of dropping a fully loaded fuel shipping cask into the cask pit or on the fuel pool floor are not discussed in this Licensing Amendment Request since Millstone Unit No. 3 is not currently licensed to transport a cask into the spent fuel building. Therefore, this event has not been included in the list of analyzed accidents associated with this licensing amendment request.

10CFR55a(a)(3)(i) Request

In accordance with 10CFR50.55a(a)(3)(i), NNECO is informing the NRC of the use of the 1995 Edition of ASME Section III Subsection NF for the design, materials, fabrication, and examination of the proposed new spent fuel storage racks, to be installed in the Millstone Point Unit 3 spent fuel pool, as an alternative to the requirements of 10CFR50.55a(b)(l).

Both the original spent fuel storage racks and the proposed new racks meet the requirements of USNRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978, and as amended January 18, 1979.

NNECO's code reconciliation evaluation confirmed that the technical requirements for the design, materials, fabrication, and examination of the proposed new spent fuel storage racks meet and exceed the original Owner requirements and the applicable Code of Construction requirements. Therefore, it is concluded that the proposed alternative will provide an acceptable level of quality and safety for the proposed new spent fuel storage racks. The design bases for the original spent fuel storage racks remain unchanged.

Proposed Technical Specification Changes

- Technical Specification Definitions 1.40 and 1.41 are reworded to provide the definitions for the new spent fuel rack configurations.

- Technical Specification 3.9.1.2 and its associated Bases Section were revised in Amendment 158 to require a boron concentration of 1,750 ppm. This change was requested by NNECO in a letter dated November 11, 1997, which identified that a seismic event of a magnitude equal to or greater than an OBE could degrade the Boraflex in the spent fuel racks. To address this situation the required boron concentration in the spent fuel pool was increased from 800 ppm to 1,750 ppm. As discussed above, the Boraflex in the existing spent fuel pool racks will not be credited for critically control when the existing racks are designated Region 3 racks. This design requirement was committed to by NNECO in the November 11, 1997, letter. The boron concentration in the spent fuel pool will only be required during fuel movements for a dropped or misplaced assembly event. Therefore, the spent fuel pool boron concentration is being revised from requiring 1,750 to 800 ppm.

- Technical Specification Surveillance 4.9.7 is being revised to clarify that the crane interlocks and stops prevent a crane from carrying a load in excess of 2,200 lbs over the spent fuel pool versus being carried over fuel assemblies as stated in the existing surveillance. This clarification more accurately describes the present crane interlocks and stops at Millstone Unit No. 3. This proposed change continues to prohibit loads in excess of 2,200 lbs from being carried over fuel in the fuel pool.

  Additionally, Technical Specification Surveillance 4.9.7 is being expanded to allow fuel pool gates and spent fuel racks to be moved by crane under administrative controls in lieu of crane interlocks and physical stops. The administrative controls will prevent the crane from carrying the load above fuel assemblies. NNECO in a response to NUREG-0612 dated March 14, 1985, stated that when placing spent fuel racks into the spent fuel pool (which weigh more than 2,200 lbs), Millstone Unit No. 3 will utilize the new fuel handling crane and bypass its interlocks so that the crane can move over the fuel pool. Additionally, in the March 14, 1985, submittal, NNECO also stated that when moving fuel pool gates (which also weigh more than 2,200 lbs), Millstone Unit No. 3 will utilize the spent fuel bridge crane and bypass its interlocks so that the crane can move over the fuel pool. However, these kinds of evolutions will require written procedures and Shift Supervisor approval. The NRC in NUREG-1031, Supplement 2 dated September 1985, referenced the March 14, 1985, submittal and stated that the overhead heavy load handling system meets the acceptance criteria of SRP Section 9.1.5.

When using administrative controls, improper operator action could lead to a crane carrying a load greater than 2,200 lbs over fuel. However, when utilizing interlocks or physical stops to prevent movement of a load greater than 2,200 lbs over fuel, improper setting of the interlocks under administrative controls, or physical failure of an interlock, could also lead to a crane carrying a load greater than 2,200 lbs over fuel. Thus, when bypassing interlocks so that a crane can carry a load greater than 2,200 lbs over the spent fuel pool, required administrative controls shall be adequate such that the probability of carrying the load over fuel is not greater than the probability of carrying the load over fuel when depending on interlocks or stops. To drop the load onto fuel requires a double malfunction, operator error or interlock failure to bring the load over fuel, and then a crane malfunction to drop the load. If Technical Specification 3.9.7 is violated, the Technical Specification requires that the load be placed into a safe condition.

Also, proposed Technical Specification Surveillance 4.9.7 clarifies that loads that weigh less than 2,200 lbs can be moved by crane under administrative controls, in lieu of crane interlocks and physical stops. This change cannot lead to violation of Technical Specification 3.9.7 because this Technical Specification only places restrictions on loads in excess of 2,200 lbs, and does not place any requirements on loads less than 2,200 lbs.

Thus, the proposed change continues to meet the requirements of Technical Specification 3.9.7, that is it prohibits a crane from carrying a load greater than 2,200 lbs over fuel in the spent fuel pool.

- Technical Specification 3.9.13 and its associated Bases Section were revised in Amendment 158 to require actions for an Operating Basis Earthquake. This change was requested by NNECO in a letter dated November 11, 1997, which identified that a seismic event of a magnitude equal to or greater than an OBE could degrade the Boraflex in the spent fuel racks. To address this situation the actions and surveillances were included in Technical Specification 3.9.13. As discussed above, the Boraflex in the existing spent fuel pool racks will not be credited for critically control when the existing racks are designated Region 3 racks. This design requirement was committed to by NNECO in the November 11, 1997, letter. The changes to the Technical Specification include: Action b will require that immediate action be initiated to move any misplaced fuel assembly into a location for which the assembly is qualified, renumber Section 4.9.13.1 to 4.9.13.1.1 which requires appropriate documentation be reviewed to assure that fuel assemblies stored in a 4-out-of-4 storage pattern in Region 1 fuel racks meet the burnup/enrichment requirements of Figure 3.9-1 (replaces old figure), add Section 4.9.13.1.2 which requires appropriate documentation be reviewed to assure that fuel assemblies stored in Region 2 fuel racks meet the burnup/enrichment requirements of Figure 3.9-3 (new figure) and add Section 4.9.13.1.3 which requires appropriate documentation be reviewed to assure that fuel assemblies stored in Region 3 fuel racks meet the burnup/enrichment/decay time requirements of Figure 3.9-4 (new figure).

- Technical Specification 3.9.14 is revised to replace the roman numeral I with the number 1 for Region 1 designation. Note, for simplicity and clarity the fuel storage region designation is being changed from roman numerals to standard numbers. This change is editorial in nature, and does not impact the rerack project design or safety.

- Technical Specification Figures 3.9-1 and 3.9-2 are replaced with new figures 3.9-1, 3.9-2, 3.9-3 and 3.9-4 indicating storage requirements for the proposed Regions 1, 2 and 3 fuel racks.

- Technical Specification Bases Section 3/4.9.1.1: BASES is revised to correct the section designator from 3/4.9.1 to 3/4.9.1.1.

- Technical Specification Bases Section 3/4.9.14: BASES is revised to recognize that Region 1 can now be either in a 3-OUT-OF-4, or 4-OUT-OF-4 storage configuration.

- Technical Specification Section 5.6.1.1: DESIGN FEATURES - CRITICALITY, is revised to describe the pitch, neutron absorber, storage pattern, and burnup/enrichment/decay time limits for each region of proposed fuel racks.

- Technical Specification Section 5.6.3: DESIGN FEATURES - CAPACITY, is revised to list the storage capacity of each proposed region of fuel racks.

- Revise INDEX pages xii and xv for new figures and page numbers.

Radiological Consequences

Radiological consequences of accidents in the spent fuel pool building have been evaluated. The existing design basis fuel drop accident in the fuel building described in FSAR Chapter 15.7.4 (fuel assembly drop onto another fuel assembly) is not affected by the rerack. Thus, potential radiological consequences from a fuel drop accident are not affected by the rerack.

A rack drop accident with radiological consequences is unlikely since all rack movement during installation will follow safe load paths that prevent heavy loads from being transported over the stored spent fuel. Thus, there are no radiological consequences from this accident.

Special Circumstance Regarding Transitioning to Revised Technical Specifications

A special circumstance will exist regarding transitioning to the proposed Technical Specifications after NRC approval of this licensing amendment request except for Technical Specification 3/4.9.7 which will take immediate effect since it does not directly deal with criticality requirements. The existing Technical Specifications credit Boraflex in the existing spent fuel racks, which reduces fuel burnup requirements. The proposed Technical Specifications eliminate Boraflex credit in the existing fuel storage racks, which causes a significant step increase in the fuel burnup requirements to store fuel in the these racks. At the time of the rerack it is anticipated that about 120 fuel assemblies stored in the Boraflex racks would not meet fuel burnup requirements of the proposed Technical Specifications. These 120 or so fuel assemblies will need to be transferred from the existing racks (called Region 3 under the proposed Technical Specifications) to the proposed additional storage racks (called Region 1 or 2 under proposed Technical Specifications) to comply with the new proposed Technical Specifications fuel burnup requirements. This means that Boraflex must be credited and existing surveillance requirements maintained until the rerack is complete, and these approximately 120 fuel assemblies can be transferred to Region 1 or Region 2 storage racks If the proposed Technical Specifications, which do not credit Boraflex, are made fully effective before NNECO can transfer these fuel assemblies out of the existing racks, the plant would not be in compliance with the revised Technical Specifications

To address this situation, NNECO proposes the following:

- When the NRC issues the rerack license amendment, NNECO would rerack the fuel pool. After rack installation and survey are complete, and as the last step of the re-rack, NNECO would transfer the approximately 120 fuel assemblies discussed above to the new Region 1 or Region 2 fuel storage racks NNECO would then fully implement the revised Technical Specifications from the rerack license amendment.

- During the interim period from NRC approval of the proposed Technical Specifications to completion of the rerack, including assembly transfer out of existing racks, NNECO will continue to comply with the existing rack Technical Specifications requirements (except for Technical Specification 3/4.9.7 which will take immediate effect). Thus, all existing Boraflex related Technical Specification requirements would remain in place until all of these approximately 120 fuel assemblies are transferred from the existing racks.

- When these approximately 120 fuel assemblies are in the process of being transferred to new racks, NNECO will administratively comply with the fuel burnup/enrichment requirements for the new racks (Regions 1 and 2) while simultaneously complying with the soluble boron requirements and Boraflex related surveillances of the existing Technical Specifications. The existing soluble boron requirements and Boraflex related surveillances are more restrictive than the proposed Technical Specifications. These actions will ensure that k_{eff} remains less than or equal to 0.95 for fuel in existing racks during the rerack, and for fuel in all racks during fuel transfer.

Docket No. 50-423
B17343

Attachment 4

Millstone Nuclear Power Station, Unit No. 3
Spent Fuel Pool Rerack (TSCR 3-22-98)

Significant Hazards Consideration and Environmental Considerations

March 1999

Significant Hazards Consideration

In accordance with 10CFR50.92, NNECO has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve a significant hazard because they would not;

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

In the analysis of the safety issues concerning the expanded pool storage capacity, NNECO has considered the following potential accident scenarios:

a. A spent fuel assembly drop with control rod and handling tool

b. A fuel pool gate drop

c. Potential damage due to a seismic event

d. Fuel assembly misloading/drop or pool temperature exceeding 160°F

e. An accidental drop of a rack module during installation activity in the pool

The probability that any of the first four accidents in the above list can occur is not significantly increased by the modification itself. All work in the pool area will be controlled and performed in strict accordance with the specific written procedures. As for an installation accident, safe load paths will be established that will prevent heavy loads from being transported over the spent fuel. Proper functioning of the cranes will be checked and verified before rack installation, and appropriate administrative controls imposed. All lift rigging and the crane/hoist system will be verified to comply with applicable plant and site procedures. All heavy lifts will be performed in accordance with established station procedures, which will comply with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." These actions will minimize the possibility of a heavy load drop accident. Fuel assembly handling procedures and techniques are not affected by adding spent fuel racks, and the probability of a fuel handling accident or misloading is not increased.

Accordingly, the proposed modification does not involve a significant increase in the probability of an accident previously evaluated.

NNECO has evaluated the consequences of an accidental drop of a fuel assembly in the spent fuel pool. The results show that such an accident will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin, k_{eff} less than or equal to 0.95, will be maintained. The radiological consequences of a fuel assembly drop are not increased from the existing postulated fuel drop accident in Millstone Unit No. 3 FSAR Section 15.7.4. Thus, the consequences of such an accident remain acceptable, and are not different from any previously evaluated accidents that the NRC has reviewed and accepted.

The consequences of an accidental drop of a fuel pool gate onto racks has been evaluated. The results show that such an accident will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin, k_{eff} less than or equal to 0.95, will be maintained. In addition, the Technical Specifications do not allow fuel to be under a fuel pool gate when one is moved. The analysis indicates no radiological consequences from this postu'a*ed accident. Thus, the consequences of such an accident remain acceptable, and are not different from any previously evaluated accidents that the NRC has reviewed and accepted.

The consequences of a design basis seismic event have been evaluated and found acceptable. The proposed additional racks and existing racks have been analyzed in their new configuration and found safe and impact-free during seismic motion, save for the baseplate-to-baseplate impacts of the proposed additional racks which are shown to cause no damage to the racks cells or Boral. The structural capability of the pool walls and basemat will not be exceeded under the loads. Thus, the consequences of a seismic event are not significantly increased.

The criticality consequences of a misloading/drop of a fuel assembly during fuel movement have been evaluated. The minimum subcriticality margin, k_{eff} less than or equal to 0.95, will continue to be maintained because of the proposed pool water soluble boron related requirements. Thus, the consequences of such an accident remain acceptable, and are not different from any previously evaluated accidents that the NRC has reviewed and accepted.

The consequences of an accidental drop of a rack module into the pool during placement have been evaluated. The analysis confirmed that very limited damage to the liner could occur, which is repairable. Any small seepage occurring is well within makeup capability, and is mitigated by emergency operating procedures. All movements of racks over the pool will comply with the applicable guidelines. Therefore, the consequences of an installation accident are not increased from any previously evaluated accident.

The consequences of a spent fuel cask drop into the pool have not been considered in this submittal since NNECO is not currently licensed to move a fuel cask into the Millstone Unit No. 3 cask pit area.

Therefore, it is concluded that the proposed changes to the Technical Specifications and licensing basis of Millstone Unit No. 3 do not significantly increase the probability or consequences of any accident previously evaluated.

2.2 Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed change does not alter the operating requirements of the plant or of the equipment credited in the mitigation of the design basis accidents. Therefore, the potential for an unanalyzed accident is not created. The postulated failure modes associated with the change do not significantly decrease the coolability, criticality margin, or structural integrity of the spent fuel in the pool. The resulting structural, thermal, and seismic loads are acceptable.

Therefore, the change does not create the possibility of a new or different kind of accident from any previously analyzed.

2.3 Involve a significant reduction in the margin of safety.

The function of the spent fuel pool is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake, fuel assembly drop, fuel pool gate drop, or drop of another heavy object. The new rack design must meet all applicable requirements for safe storage and be functionally compatible with the other rack design in the spent fuel pool.

NNECO has addressed the safety issues related to the expanded pool storage capacity in the following areas:

1. Material, mechanical, and structural considerations

2. Nuclear criticality

3. Thermal-hydraulic and pool cooling

The mechanical, material, and structural designs of the new racks have been reviewed in accordance with the applicable provisions of NRC "OT Position for the Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978, as amended January 18, 1979. The rack materials used are compatible with the spent fuel assemblies and the spent fuel pool environment. The design of the new racks preserves the proper margin of safety during abnormal loads such as a dropped fuel assembly, a postulated seismic event, a dropped fuel pool gate, and tensile loads from a stuck fuel assembly. It has been shown that such loads will not invalidate the mechanical design and material selection to safely store fuel in a coolable and subcritical configuration. Also, it has been shown that the pool structure will maintain its integrity and function during normal operation, all postulated accident sequences, and postulated seismic events.

The methodology used in the criticality analysis of the expanded spent fuel pool storage capacity meets the appropriate NRC guidelines and the ANSI standards. The margin of safety for subcriticality is determined by a neutron multiplication factor less than or equal to 0.95 under all accident conditions, including uncertainties. This criterion has been preserved in all analyzed accidents and seismic events.

The special circumstance regarding transitioning to the revised technical specifications was discussed. At present, NNECO estimates that there will be approximately 120 fuel assemblies stored in existing racks that will not meet the burnup/enrichment requirements for storage in these racks under the proposed Technical Specifications. During the actual reracking effort, including transfer of these assemblies from existing racks to Region 1 and 2 racks, existing soluble boron and Boraflex related requirements and surveillances will continue to be enforced. Also, when transferring these assemblies to Region 1 and 2 racks, the burnup/enrichment requirements of these racks will be enforced. After fuel transfer is complete, the revised Technical Specifications will be fully implemented. These requirements ensure that the neutron multiplication factor will remain less than or equal to 0.95 during the whole period of the rerack.

The rerack thermal hydraulic analysis is based on NNECO's January 18, 1999, submittal analysis which bounds the heat load of this licensing amendment request. The rerack thermal hydraulic analysis found that, in the blocked hottest stored assembly, the local peak water temperature will remain below boiling, and the fuel clad will not experience high temperatures.

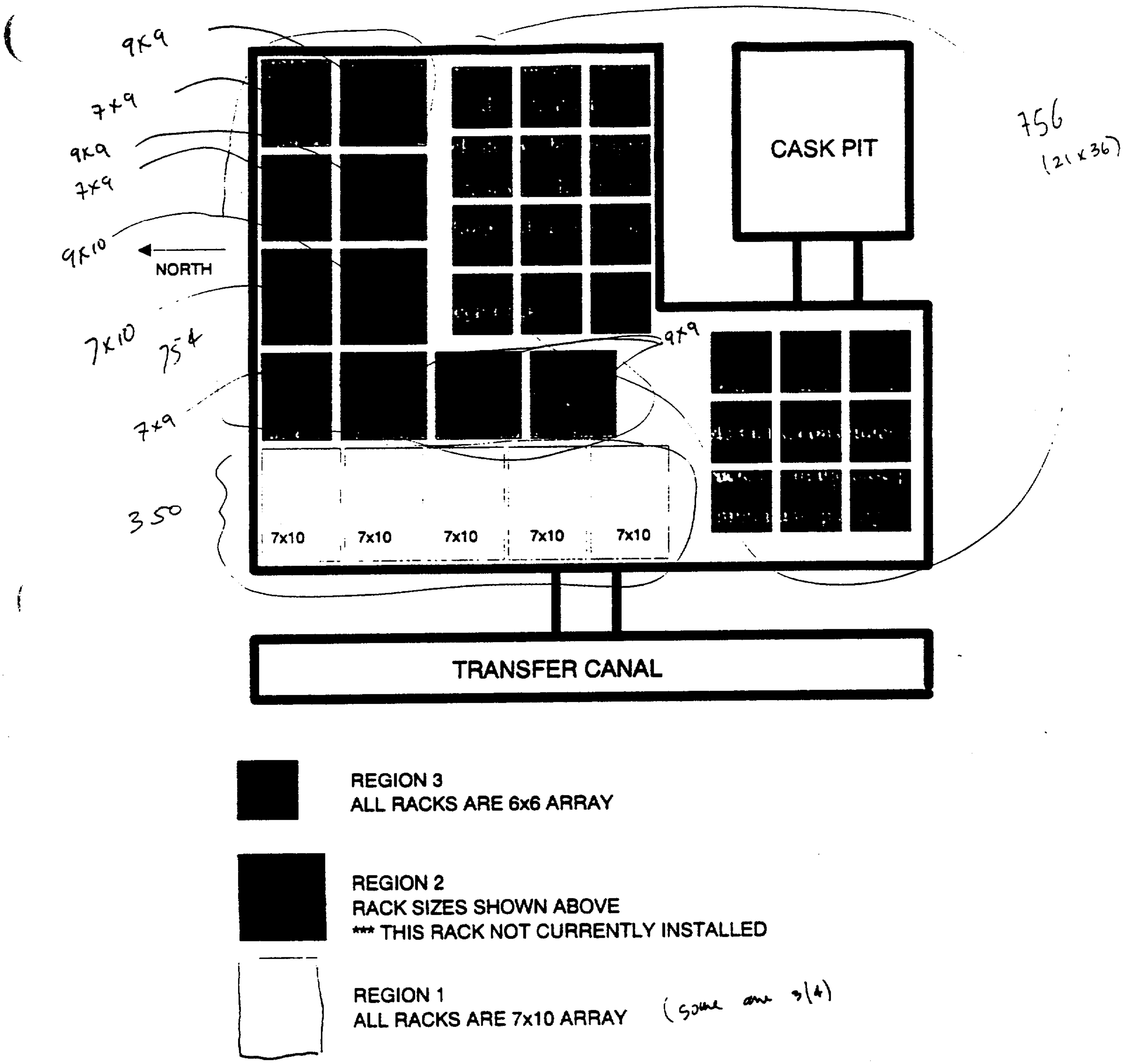
Regarding Technical Specification Surveillance 4.9.7, since the proposed change continues to meet the requirements of Technical Specification 3.9.7, that is it prohibits a crane from carrying a load greater than 2,200 lbs over fuel in the spent fuel pool to preclude fuel damage, the margin of safety is maintained.

Thus, it is concluded that the proposed changes to the Technical Specifications and licensing basis of Millstone Unit No. 3 do not involve a significant reduction in the margin of safety at Millstone Unit No. 3.

Environmental Considerations

NNECO has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed revision does not involve a significant hazard, does not significantly increase the type and amounts of effluents that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, NNECO concludes that the proposed revision meets the criteria delineated in 10CFR51.22(c)(9) for categorical exclusion from the requirements for environmental review.

MILLSTONE UNIT 3 SPENT FUEL POOL LAYOUT



(NOT DRAWN TO SCALE)

FIGURE 1

Docket No. 50-423
B17343

Attachment 5

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specification
Spent Fuel Pool Rerack (TSCR 3-22-98)

Non-Proprietary Version of
Licensing Report for Spent Fuel Rack Installation
at Millstone Nuclear Station Unit 3

March 1999

LICENSING REPORT

for

SPENT FUEL RACK INSTALLATION

at

MILLSTONE NUCLEAR STATION
UNIT 3

Table of Contents

1.0 INTRODUCTION

Millstone Point Unit 3 (MP3) is a Westinghouse Pressurized Water Reactor (PWR) owned and operated by Northeast Utilities (NU). The plant is located at a three unit site in the town of Waterford, Connecticut. A license was granted by the USNRC on January 31, 1986 and full commercial operation of the 1150 MWe plant began in 1986.

The MP3 reactor core contains 193 assemblies. During re-fueling, spent fuel is placed in the plant's pool; an L - shaped basin with a total nominal area of approximately 1,574 ft^2. The pool presently contains 756 storage cells which were installed during original plant construction. The twenty-one existing storage racks are of end-connected-construction (ECC). Each contains a 6 x 6 array. As is true for all ECC racks, the individual boxes are connected to each other at their extremities; there is no longitudinal inter-cell connection between the cells. The ECC racks employ a 0.06 inch wall storage cell at a pitch of 10.35 inches, with Boraflex serving as the neutron absorber.

This license application addresses installation of fifteen high-density racks in the MP3 pool. These fifteen high density racks have a maximum capacity of 1,104 storage cells. Additional storage capacity is needed since MP3 will lose its full-core reserve discharge capacity at the end of its seventh cycle. Tables 1.1 and 1.2 demonstrate this. Table 1.1 shows the historic and projected discharges into the MP3 pool. Table 1.2 shows the current and post-modification storage capacities. The new racks will extend the date of loss-of-full-core-reserve discharge capability approaching end of license (see Table 1.2).

Northeast Utilities plans to install fourteen modules initially and the fifteenth rack (A5) at a later date. The analyses include the fifteenth rack. Figure 2.1 of Section 2 shows the planned layout. The existing fuel racks will not be moved. However, credit for Boraflex as a neutron absorber will be eliminated.

The new high density racks proposed for MP3 have been designed by Holtec International of Marlton, New Jersey. The racks are free-standing and self-supporting. The principal construction materials for the new racks are ASME SA240-Type 304L stainless steel sheet and plate stock and SA564 (precipitation hardened stainless steel for the adjustable support spindles). The only non-stainless material utilized in the rack is the neutron absorber material, which is a boron carbide aluminum cermet manufactured under a U.S. patent and sold under the brand name Boral™ by AAR Advan[illegible] [illegible] [illegible]vonia, Michigan.

The new rac[illegible] [illegible] [illegible]esi[illegible] [illegible]d analyzed in accordance with Section III, Division 1, Subsection NF of the ASM[illegible] [illegible] [illegible] [illegible]ssure Vessel Code. The material procurement and fabrication of the rack modules [illegible] [illegible] CFR50 Appendix B requirements. The racks proposed for the MP3 pool are iden[illegible] [illegible] the[illegible] [illegible]atomical details to racks recently provided by Holtec International to many PWR plants. Table 1.3 lists recently licensed PWR plants with racks similar to those proposed for M[illegible]

This Licensing Report documents the design and analyses performed to demonstrate that the new spent fuel racks satisfy all requirements of the governing codes and standards The safety assessment of the proposed rack modules involves demonstration of thermal-hydraulic, criticality, and structural adequacy. Thermal-hydraulic adequacy requires that the fuel cladding withstand the imposed thermal stress and that the steady state bulk pool temperature remain within prescribed limits. The criticality analyses show that the neutron multiplication factor (keff) for the stored fuel array is bounded by the MP3 limit of 0.945 (the USNRC limit is 0.95) under assumptions of 95% probability and 95% confidence. Consequences of inadvertent placement of a fuel assembly are also evaluated as part of the criticality analysis. The demonstration of structural adequacy of the rack modules shows that the free-standing modules and pool walls maintain the stored fuel within the configurati[illegible] [illegible]sidered in the thermal-hydraulic and criticality analysis under all load conditions.

This document has been prepared for submission to the U.S. Nuclear Regulatory Commission for securing regulatory approval of the modification of the MP3 pool as proposed herein.

Table 1.1 MP3 HISTORIC AND PROJECTED FUEL DISCHARGE SCHEDULE			
End-of-Cycle Discharged	Bundles Permanently Discharged	Total Number of Fuel Assemblies	Date Discharged
1	75	75	10/87
2	85	160	5/89
3	79	239	2/91
4	68	307	7/93
5	109	416	4/95
6	85	501	3/99
7	84	585*	11/00
8	85	670	9/02
9	84	754	6/04
10	85	839	4/06
11	84	923	2/08
12	85	1,008	11/09
13	84	1,092	9/11
14	85	1,177	7/13
15	84	1,261	4/15
16	85	1,346	2/17
17	84	1,430	12/18
18	85	1,515	9/20
19	84	1,599	7/22
20	85	1,684**	5/24
21	193	1,877	2/26

* Loss of Full-Core-Reserve with current storage capacity

** Loss of Full-Core-Reserve with new racks installed

Table 1.2

AVAILABLE STORAGE IN MP3 POOL AT PRESENT AND AFTER CAMPAIGN I EXTENSION

Refueling No.	Refueling Outage Date	Discharge Size	With Present Available Capacity (756 Cells)	[redacted]
6	3/99	85	255	[redacted]
7	11/00	84	171*	[redacted]
8	9/02	85	86	[redacted]
9	6/04	84	2	[redacted]
10	4/06	85		[redacted]
11	2/08	84		[redacted]
12	11/09	85		[redacted]
13	9/11	84		[redacted]
14	7/13	85		[redacted]
15	4/15	84		[redacted]
16	2/17	85		[redacted]
17	12/18	84		[redacted]
18	9/20	85		[redacted]
19	7/22	84		[redacted]
20	5/24	85		[redacted]
21	2/26	193		[redacted]

* Loss of Full-Core-Reserve with current storage capacity
** Loss of Full-Core-Reserve with new racks installed

Table 1.3

PRESENTLY LICENSED PEER SITES WITH RACK DESIGNS SIMILAR TO THAT IN THIS APPLICATION

Plant	Docket Number	Year Licensed
Sequoyah	50-327 50-328	1994
Connecticut Yankee	50-213	1994
Fort Calhoun	50-285	1994
Salem 1 & 2	50-272 50-311	1994
Beaver Valley	50-334	1992
D. C. Cook	50-315 50-316	1992
Zion	50-295 50-304	1992
Three Mile Island 1	50-289	1990

2.0 OVERVIEW OF PROPOSED CAPACITY EXPANSION

2.1 General Description

This section provides general information on the new storage modules proposed for the MP3 spent fuel pool. It also describes the basis for the detailed criticality, thermal-hydraulic, and seismic analyses presented in subsequent sections of this report.

The storage capacity expansion of the MP3 spent fuel pool features a two region arrangement. In the proposed scheme, a group of five modules will store the most reactive fuel (up to 5 weight % by volume (w/o)) without any burnup limitation in a 3-out-of-4 configuration, with the fourth location blocked and empty of fuel. Fuel may be stored in these racks in a 4-out-of-4 configuration with an enrichment/burnup limitation. These racks will use a flux-trap design. The grouping of flux-trap racks is referred to as Region 1. The remaining ten racks do not use flux-traps and are collectively referred to as Region 2. Region 2 racks have an enrichment/burnup limitation on them. Figure 2.1 shows the module layout.

The existing spent fuel storage racks are collectively referred to as Region 3. The existing racks are not moved or modified in any way by this rerack. As discussed in Section 4 of this report, the Region 3 racks will no longer credit Boraflex as a neutron absorber material.

Table 2.1 provides geometric and physical data for Region 1 and Region 2 cells. The rack modules have five distinct sizes, denoted as types A, B, C, D, and E. Table 2.2 gives the number of cells in each of these rack types. As indicated in the table, the rerack would provide an additional 1,104 storage locations. The module dimensions and weights are presented in Table 2.3.

The proposed modules for the MP3 fuel pool are qualified as freestanding racks.

2.2 Design Basis

This section describes the concepts and features that underlie the design of the new MP3 rack modules. The key criteria are set forth in the classical USNRC memorandum entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978 as modified by amendment dated January 18, 1979. The individual sections of this report expound on the specific design bases derived from the above-mentioned "OT Position Paper". Nevertheless, a brief summary of the design bases for the MP3 racks are summarized in the following:

a. Disposition: All new rack modules are required to be free-standing.

b. Kinematic Stability: All free-standing modules must be kinematically stable (against tipping or overturning) when a seismic event that is 150% of the postulated SSE is imposed.

c. Structural Compliance: All primary stresses in the rack modules must satisfy the limits postulated in Section III, subsection NF of the ASME Boiler and Pressure Vessel Code.

d. Thermal-Hydraulic Compliance: The spatial average bulk pool temperature is required to remain under 150°F in the wake of a normal refueling with single active failure of one train of spent fuel pool cooling. In addition to the limitations on the bulk pool temperature, the local water temperature in the MP3 pool must remain subcooled (i.e., below the boiling temperature coincident with local elevated pressure conditions).

e. Criticality Compliance: The flux-trap storage cells (Region 1) must be able to store fresh Zircaloy clad fuel with 5 w/o initial enrichment in a 3-out-of-4 configurtion while maintaining the reactivity ≤ 0.945. Region 2 cells must be able to store the Zircaloy clad fuel of 5 w/o enrichment and 39,000 MWD/MTU burnup while maintaining the reactivity ≤ 0.945.

f. Radiological Compliance: The reracking of Millstone 3 must not lead to violation of the off-site dose limits, or adversely affect the area dose environment as set forth in the Millstone Unit 3 FSAR.

g. Pool Structure: The ability of the reinforced concrete structure to satisfy the load combinations set forth in NUREG-0800. SRP 3.8.4 must be demonstrated.

h. Rack Stress Fatigue: In addition to satisfying the primary stress criteria of Subsection

NF, the alternating local stresses in the rack structure during a seismic event are also required to be sufficiently bounded such that the "cumulative damage factor" due to one SSE and five OBE events does not exceed 1.0.

i. Liner Integrity: The integrity of the liner under cyclic in-plane loading during a seismic event must be demonstrated.

j. Bearing Pads: The bearing pads must be sufficiently thick such that the pressure on the liner continues to satisfy the ACI limits during and after a design basis seismic event.

k. Accident Events: In the event of postulated drop events (uncontrolled lowering of a fuel assembly, for instance), it is necessary to demonstrate that the subcriticality of the rack structure and its thermal hydraulic adequacy are not compromised.

l. Construction Events: The field construction services required to be carried out for executing the reracking must be demonstrated to be within the "state of proven art".

The foregoing design bases are further articulated in subsequent sections of this licensing report.

2.3 Codes, Standards, and Practices for the Spent Fuel Pool Modification

The design and fabrication of the rack modules is performed under a strict quality assurance program which meets 10CFR50 Appendix B requirements.

The following codes, standards and practices are used for all applicable aspects of the design, construction, and assembly of the spent fuel storage racks. Additional specific references related to detailed analyses are given in each section.

a. Design Codes

1. AISC Manual of Steel Construction, 8th Edition, 1980 (provides detailed structural criteria for linear type supports).

2. ANSI N210-1976, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations" (contains guidelines for fuel rack design).

3. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 1, 1995 Edition.

4. ANSI/AISC-N690-1984 - Nuclear Facilities - Steel Safety Related Structure for Design, Fabrication and Erection.

5. ASNT-TC-1A, 1984 American Society for Nondestructive Testing (Recommended Practice for Personnel Qualifications).

6. ACI 349-85 - Code Requirements for Nuclear Safety Related Concrete Structures.

b. Material Codes - Standards of ASME or ASTM, as noted:

1. ASME SA240 - Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels.

2. ASTM A262 - Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel.

3. ASME SA276 - Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes.

4. ASME SA479 - Steel Bars for Boilers & Pressure Vessels.

5. ASTM C750 - Standard Specification for Nuclear-Grade Boron Carbide Powder.

6. ASTM C992 - Standard Specification for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Spent Fuel Storage Racks.

7. ASME SA312 - Specification for Seamless and Welded Austenitic Stainless Steel Pipe.

8. ASME SA564 - Specification for Hot Rolled and Cold-Finished Age-Hardening Stainless and Heat Resisting Steel Bars and Shapes.

9. American Society of Mechanical Engineers (ASME). Boiler and Pressure Vessel Code, Section II-Parts A and C, 1995.

10. ASTM A262 Practices A and E - Standard Recommended Practices for Detecting Susceptibility to Intergrannular Attack in Stainless Steels.

11. ASTM A380 - Recommended Practice for Descaling, Cleaning and Marking Stainless Steel Parts and Equipment.

c. Welding Codes

1. ASME Boiler and Pressure Vessel Code, Section IX -Welding and Brazing Qualifications, 1995.

2. AWS D1.1 - Welding Standards (1989).

d. Quality Assurance, Cleanliness, Packaging, Shipping, Receiving, Storage, and Handling Requirements

1. NQA-2-Part 2.2 1983 - Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants (During Construction Phase).

2. NQA-1-1983 - Basic Requirements and Supplements.

3. ASME Boiler and Pressure Vessel, Section V, Nondestructive Examination, 1995 Edition.

4. ANSI - N45.2.6 - Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants (Regulatory Guide 1.58).

e. Governing NRC Design Documents

1. "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and the modifications to this document of January 18, 1979.

2. NRC Resolution of Generic Technical Activity A-36, July 1980, NUREG-0612, Control of Heavy Loads in Nuclear Power Plants.

f. Other ANSI Standards (not listed in the preceding)

1. ANSI/ANS 8.1 - 1983, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.

2. ANSI/ANS 8.7 - 1974, Guide for Nuclear Criticality Safety in the Storage of Fissile Materials.

3. ANSI/ANS 8.11 - 1975, Validation of Calculation Methods for Nuclear Criticality Safety.

g. Code-of-Federal Regulations

1. 10CFR21 - Reporting of Defects and Non-compliance.

2. 10CFR50 - Appendix A - General Design Criteria for Nuclear Power Plants.

3. 10CFR50 - Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

4. 10CFR Part 20 - Radiation Protection Standards.

5. 29CFR Section 1910.401 - OSHA Standards for Commercial Diving Operations.

h. Regulatory Guides

1. RG 1.13 - Spent Fuel Storage Facility Design Basis.

2. RG 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility of Boiling and Pressurized Water Reactors.

3. RG 1.28 - (endorses ANSI N45.2) - Quality Assurance Program Requirements, June, 1972.

4. RG 1.29 - Seismic Design Classification.

5. RG 1.38 - (endorses ANSI N45.2.2) Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants, March, 1973.

6. RG 1.44 - Control of the Use of Sensitized Stainless Steel.

7. RG 1.58 - (endorses ANSI N45.2.6) Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel. Rev. 1, September, 1980.

8. RG 1.64 - (endorses ANSI N45.2.11) Quality Assurance Requirements for the Design of Nuclear Power Plants, October, 1973.

9. RG 1.74 - (endorses ANSI N45.2.10) Quality Assurance Terms and Definitions, February, 1974.

10. RG 1.88 - (endorses ANSI N45.2.9) Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records. Rev. 2, October, 1976.

11. RG 1.92 - Combining Modal Responses and Spatial Components in Seismic

Response Analysis.

12. RG 1.123 - (endorses ANSI N45.2.13) Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants.

13. NRC Regulatory Guide 3.41 Rev., May 1977 - Validation of Calculation Methods for Nuclear Criticality Safety.

14. NRC Regulatory Guide 1.26 Rev. 3, Feb. 1976, Quality Group Classifications and Standards for Water, Steam and Radioactive Containing Components of Nuclear Power Plants.

i. Branch Technical Position

1. CPB 9.1-1 - Criticality in Fuel Storage Facilities.

2. ASB 9-2 - Residual Decay Energy for Light-Water Reactors for Long-Term Cooling.

j. Standard Review Plan (NUREG-0800, July 1981)

1. SRP 3.7.1 - Seismic Design Parameters.

2. SRP 3.7.2 - Seismic System Analysis.

3. SRP 3.7.2 - Seismic Subsystem Analysis.

4. SRP 3.8.4 - Other Seismic Category I Structures (including Appendix D).

5. SRP 9.1.2 - Spent Fuel Storage.

6. SRP 9.1.3 - Spent Fuel Pool Cooling and Cleanup System.

k. Other

MP3 Final Safety Analysis Report (FSAR).

MP3 Technical Specification.

NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment", April 11, 1996.

2.4 Quality Assurance Program

The governing quality assurance requirements for fabrication of the MP3 spent fuel racks are enunciated in 10CFR50 Appendix B. The quality assurance program for design of the Millstone Unit 3 racks are described in Holtec's Nuclear Quality Assurance Manual, which has been reviewed and approved by Northeast Utilities (NU). This program is designed to provide a flexible but highly controlled system for the design, analysis and licensing of customized components in accordance with various codes, specifications, and regulatory requirements.

The manufacturing of the racks will be carried out by Holtec's designated manufacturer (U.S. Tool & Die, Inc.). The Quality Assurance System enforced on the manufacturer's shop floor shall provide for all controls necessary to fulfill all quality assurance requirements with sufficient simplicity to make it functional on a day-to-day basis. UST&D has manufactured high density racks for over 60 nuclear plants around the world. UST&D has been audited by the industry group NUPIC, and the QA branch of NMSS with most satisfactory results.

The Quality Assurance System that will be used by Holtec to install the racks is also controlled by the Holtec Nuclear Quality Assurance Manual and by NU's site-specific requirements.

2.5 Mechanical Design

The Millstone Unit 3 rack modules are designed as cellular structures such that each fuel cell has a prismatic square opening with conformal lateral support and a flat horizontal bearing surface.

Figures 2.2 and 2.3 show pictorial views of Region 1 and Region 2 modules, respectively. As can be inferred from these schematic representations, the high density modules for MP3 have been designed to simulate multi-flange beam structures resulting in excellent detuning characteristics with respect to the applicable seismic events.

Figure 2.4 provides an elevation view of the Region 1 and Region 2 racks located in the Spent Fuel Pool.

Table 2.1

DESIGN DATA FOR NEW RACKS

...ments. The key criteria are set forth in the USNRC memorandum entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978 as modified by amendment dated January 18, 1979. The individual sections of this report expound on the specific design bases derived from the above-mentioned "OT Position Paper". Nevertheless, a brief summary of the design bases for the MP3 racks are summarized in the following:

a. Disposition: All new rack modules are required to be free-standing.

b. Kinematic Stability: All free-standing modules must be kinematically stable (against tipping or overturning) when a seismic event that is 150% of the postulated SSE is imposed.

c. Structural Compliance: All primary stresses in the rack modules must satisfy the limits postulated in Section III, subsection NF of the ASME Boiler and Pressure

Table 2.2

MODULE DATA FOR RERACK CAMPAIGN I

MODULE		NUMBER OF CELLS			
I.D.	QTY.	North-South Direction	East-West Direction	Total Per Rack	Total No. of Cells for this Rack Type
A	5	9	9	81	405
B	1	9	10	90	90
C	1	7	10	70	70
D	5	7	10	70	350
E	3	7	9	63	189
TOTAL:	**15**	--	--	----	**1,104**

Table 2.3

MODULE DIMENSIONS AND WEIGHTS FOR RERACK CAMPAIGN I

Module I.D.	Dimension (inches)[†]		Shipping Weight in Pounds
	North-South	East-West	
A	81.53	81.53	13,090
B	81.53	90.54	14,410
C	63.49	90.54	11,490
D	69.33	103.38	18,085
E	63.49	81.53	10,450

[†] Nominal rectangular planform dimensions.

PROPRIETARY

FIGURE 2.1: POOL LAYOUT MILLSTONE UNIT No. 3
(CAMPAIGN 1 INSTALLATION)

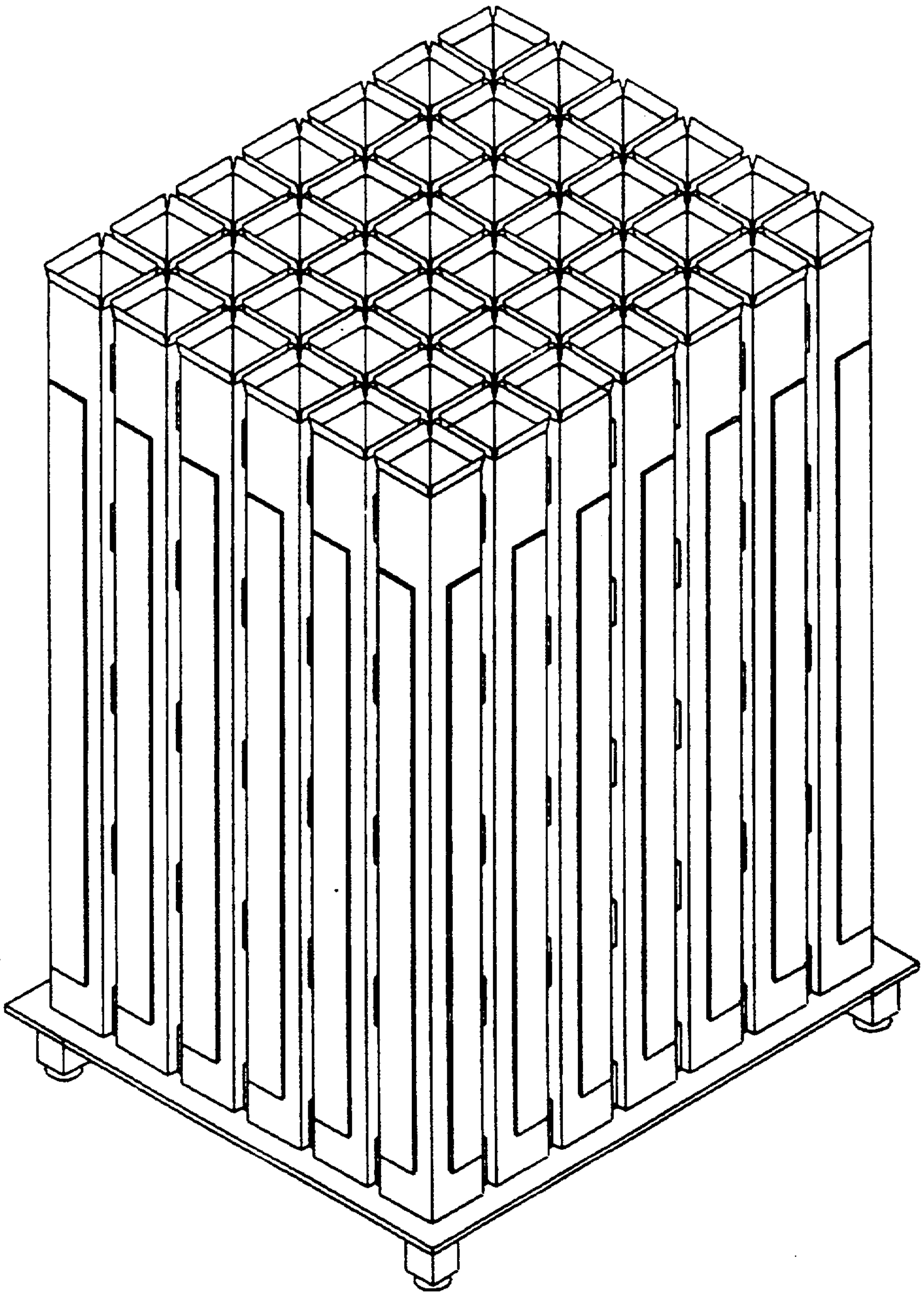


FIGURE 2.2; PICTORIAL VIEW OF REGION 1 RACK STRUCTURE

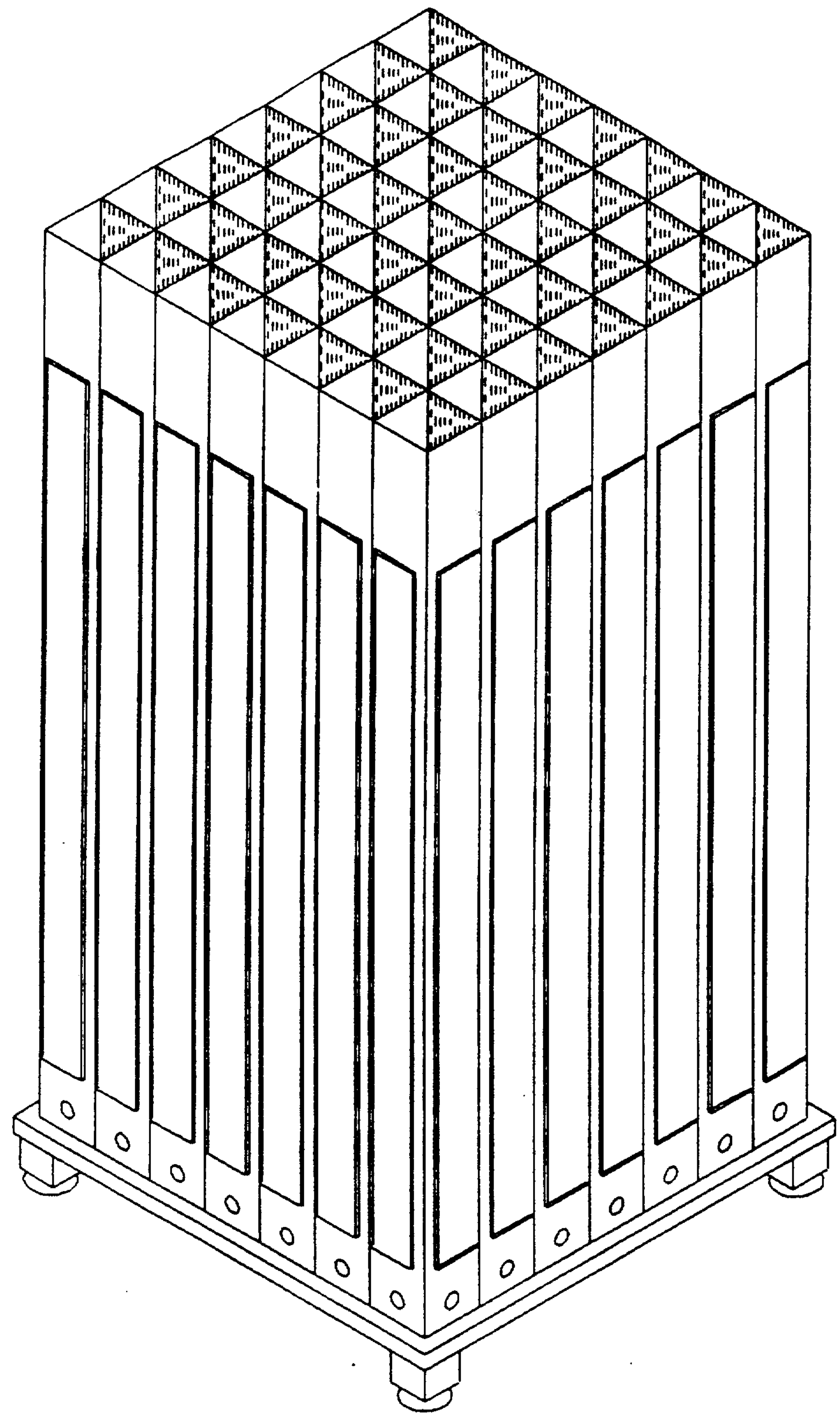
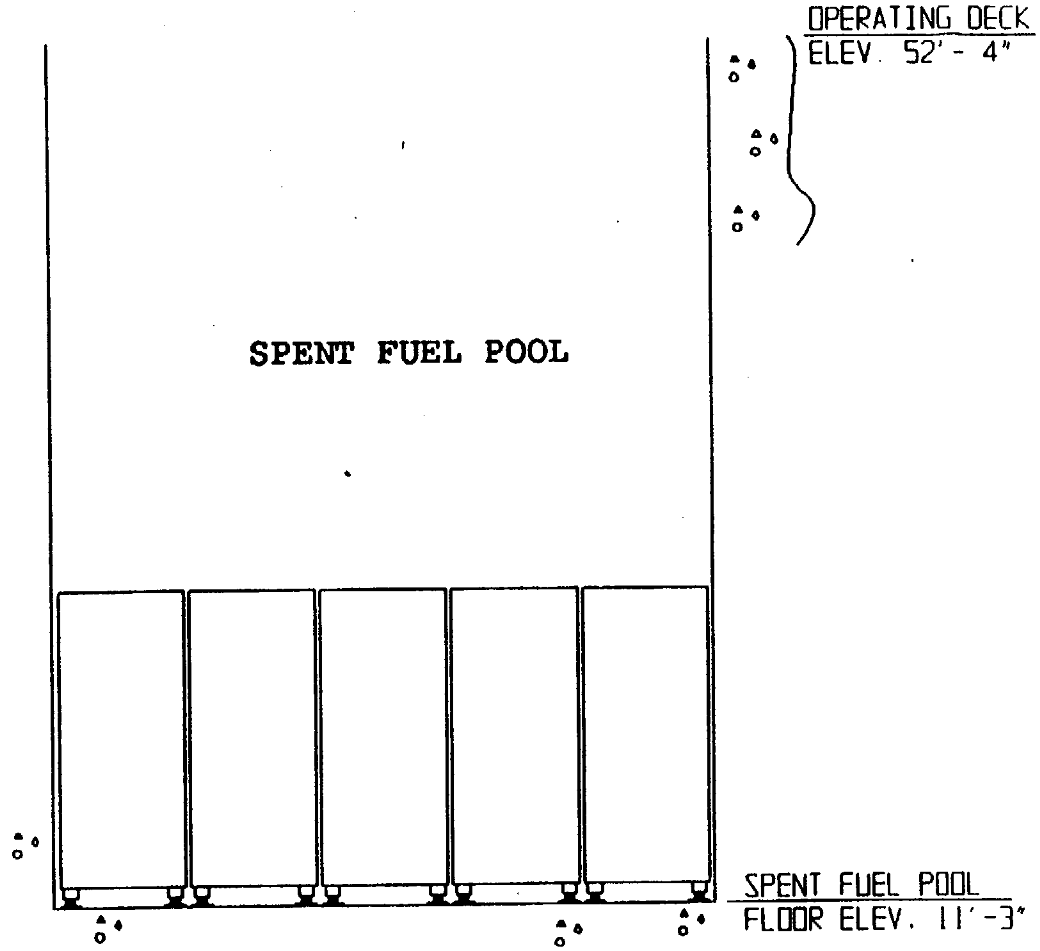


FIGURE 2.3: PICTORIAL VIEW OF REGION 2 RACK STRUCTURE



FIGUREW 2.4; ELEVATION VIEW OF RACK LAYOUT

3.0 FABRICATION, MATERIALS, AND HEAVY LOADS CONSIDERATIONS

3.1 Rack Fabrication

The object of this section is to provide a self-contained description of rack module construction and to enable an independent appraisal of the adequacy of design.

3.1.1 Fabrication Objective

The are four interrelated manufacturing requirements for MP3's high density storage racks,

1. The rack modules are fabricated in such a manner that there is no weld splatter on the storage cell surfaces which would come in contact with the fuel assembly.
2. The storage locations are constructed so that redundant flow paths for the coolant are available.
3. The fabrication process involves operational sequences which permit immediate verification by the inspection staff.
4. The storage cells are connected to each other by austenitic stainless steel corner welds which leads to a honeycomb lattice construction. The extent of welding is selected to "detune" the racks from the seismic input motion (OBE and SSE).

3.1.2 Rack Module for Region 1

This section describes the Region 1 fabrication sequence.

The rack module manufacturing begins with fabrication of the "box". The boxes are fabricated from two precision formed channels by seam welding in a machine equipped with copper chill bars and pneumatic clamps to minimize distortion due to welding heat input. Figure 3.1 shows the box.

The minimum weld penetration is [redacted] of the [redacted] inch box metal gage. A die is used to flare out one end of the box to provide the tapered lead-in (Figure 3.2). Three-quarter inch diameter holes are punched on all four sides near the other end of the box to provide the requisite auxiliary flow holes. Each box constitutes a storage location. Each external side of the box is equipped with a stainless steel sheath that holds one integral Boral sheet (poison material).

The design objective calls for attaching Boral tightly on the box surface. This is accomplished by die forming the internal and external box sheathings, as shown in Figure 3.3. The flanges of the sheathing are attached to the box using skip welds and spot welds. The sheathings serve to locate and position the poison sheet accurately, and to preclude its movement under seismic conditions.

Having fabricated the required number of composite box assemblies, they are joined together in a fixture using connector elements in the manner shown in Figure 3.4. Figure 3.5 shows an elevation view of two storage cells of a Region 1 rack module. A representative connector element is also shown in the figure. Joining the cells by the connector elements results in a well- defined shear flow path and essentially makes the box assemblage into a multi-flanged beam-type structure. The "baseplate" is attached to the bottom edge of the boxes. The baseplate is [redacted] inch thick austenitic stainless steel plate stock which has [redacted] inch diameter holes (except lift locations, which are rectangular) cut out in a pitch identical to the box pitch. The baseplate is attached to the cell assemblage by fillet welding the box edge to the plate.

In the final step, adjustable leg supports (shown in Figure 3.6) are welded to the underside of the baseplate. The adjustable legs provide a [redacted] inch vertical height adjustment at each leg location.

Appropriate NDE (nondestructive examination) occurs on all welds including visual examination of sheathing welds, box longitudinal seam welds, box-to-baseplate welds, and box-to-box connection welds; and liquid penetrant examination of support leg welds, in accordance with the design drawings.

3.1.3 Rack Module for Region 2

Region 2 storage cell locations have a single poison panel between adjacent box wall surfaces. There are five significant components (discussed below) of the Region 2 racks: (1) the storage box subassembly (2) the baseplate, (3) the neutron absorber material, (4) the sheathing, and (5) the support legs.

1. Storage cell box subassembly: As described for Region 1, the boxes are fabricated from two precision formed channels by seam welding in a machine equipped with copper chill bars and pneumatic clamps to minimize distortion due to welding heat input. Figure 3.1 shows the box.

   Each box has four lateral holes punched near its bottom edge to provide auxiliary flow holes. A sheathing is attached to each side of the box with the poison material installed in the sheathing cavity. The edges of the sheathing and the box are welded together to form a smooth edge. The box, with integrally connected sheathing, is referred to as the "composite box".

   The composite boxes are arranged in a checkerboard array to form an assemblage of storage cell locations (Figure 3.7). Filler panels and corner angles are welded to the edges of boxes at the outside boundary of the rack to make the peripheral formed cells. The inter-box welding and pitch adjustment are accomplished by small longitudinal connectors. This assemblage of box assemblies is welded edge-to-edge as shown in Figure 3.7, resulting in a honeycomb structure with axial, flexural and torsional rigidity depending on the extent of intercell welding provided. It can be seen from Figure 3.7 that two edges of each interior box are connected to the contiguous boxes resulting in a well-defined path for "shear flow".

2. Baseplate: The baseplate provides a continuous horizontal surface for supporting the fuel assemblies. The baseplate has a [redacted] inch diameter hole (except lift locations which are rectangular) in each cell location as described in the preceding section. The baseplate is attached to the cell assemblage by fillet welds.

3. The Neutron Absorber Material: As mentioned in the preceding section, Boral is used as the neutron absorber material.

4. Sheathing: As described earlier, the sheathing serves as the locator and retainer of the poison material.

5. Support legs: As stated earlier, all support legs are the adjustable type (Figure 3.6). The top position is made of austenitic steel material. The bottom part is made of 17:4 Ph series stainless steel to avoid galling problems.

   Each support leg is equipped with a readily accessible socket to enable remote leveling of the rack after its placement in the pool.

   An elevation view of three contiguous Region 2 cells is shown in Figure 3.8.

3.2 Material Considerations

3.2.1 Introduction

Safe storage of nuclear fuel requires that the materials utilized in the fabrication of racks be of proven durability and be compatible with the pool water environment. This section provides the necessary information on this subject.

3.2.2 Structural Materials

The following structural materials are utilized in the fabrication of the spent fuel racks:

a. ASME SA240-304L for all sheet metal stock.

b. Internally threaded support legs: ASME SA240-304L.

c. Externally threaded support spindle: ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100°F).

d. Weld material - per the following ASME specification: SFA 5.9 R308L.

3.2.3 Poison Material

In addition to the structural and non-structural stainless material, the racks employ Boral™, a patented product of AAR Advanced Structures, as the neutron absorber material.

Boral is a thermal neutron poison material composed of boron carbide and 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The 1100 alloy aluminum is a light-weight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal and chemical environment of a nuclear reactor or the spent fuel pool.

Boral's use in spent fuel pools as the neutron absorbing material can be attributed to the following reasons:

i. The content and placement of boron carbide provides a very high removal cross section for thermal neutrons.

ii. Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.

iii. The boron carbide and aluminum materials in Boral are totally unaffected by long-term exposure to radiation.

iv. The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.

v. Boral is stable, strong, durable, and corrosion resistant.

Holtec International's QA program ensures that Boral is manufactured by AAR Brooks & Perkins under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants".

As indicated in Table 3.1, Boral has been licensed by the USNRC for use in numerous BWR and PWR spent fuel storage racks and has been extensively used in overseas nuclear installations.

Boral Material Characteristics

Aluminum: Aluminum is a silvery-white, ductile metallic element that is the most abundant in the earth's crust. The 1100 alloy aluminum is used extensively in heat exchangers, pressure and storage tanks, chemical equipment, reflectors and sheet metal work. It has high resistance to corrosion in industrial and marine atmospheres. The physical, mechanical and chemical properties of the 1100 alloy aluminum are listed in Tables 3.2 and 3.3.

The excellent corrosion resistance of the 1100 alloy aluminum is provided by the protective oxide film that develops on its surface from exposure to the atmosphere or water. This film prevents the loss of metal from general corrosion or pitting corrosion and the film remains stable between a pH range of 4.5 to 8.5.

Boron Carbide: The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The particles range in size between 60 and 200 mesh and the material conforms to the chemical composition and properties listed in Table 3.4.

3.2.4 Compatibility with Coolant

All materials used in the construction of the MP3 racks have an established history of in-pool usage. Their physical, chemical and radiological compatibility with the pool environment has been established throughout the industry. As noted in Table 3.1, Boral has been used in both vented and unvented configurations in fuel pools with equal success. Austenitic stainless steel is the most widely used stainless alloy in nuclear power plants.

3.3 Heavy Load Considerations for the Proposed Reracking Operation

A 10-ton crane will be utilized for handling all heavy loads in the reracking operation. A remotely engageable lift rig, meeting NUREG-0612 stress criteria, will be used to lift the new modules. It consists of independently loaded lift rods with a "cam type" lift configuration. This ensures that failure of one traction rod will not result in uncontrolled lowering of the load; compliant with Section 5.1.6(1) of NUREG-0612. The remotely engageable lift rig also has the following attributes:

a. The stresses in the lift rods are self limiting inasmuch as an increase in the magnitude of the load reduces the eccentricity between the upward force and downward reaction (moment arm).

b. It is impossible for a traction rod to lose its engagement with the rig in locked position due to the load of the lifted rack pulling each traction rod in the downward direction, thus keeping it within its locking slots. Moreover, the locked configuration can be directly verified from above the pool water without the aid of an underwater camera due to the orientation of position locator flags atop each traction rod.

c. The stress analysis of the rig is carried out and the primary stress limits postulated in ANSI 14.6 (1978) are shown to be met.

d. The rig is load tested with 300% of the maximum weight to be lifted. The test weight is maintained in the air for one hour. All critical weld joints are liquid penetrant examined, after the load test, to establish the soundness of all critical joints.

Pursuant to the defense-in-depth approach of NUREG-0612, the following additional measures of safety will be undertaken for the reracking operation.

i. The cranes and lifting devices used in the project will be given a preventive maintenance checkup and inspection per the MP3 procedures before beginning the reracking operation.

ii. Safe load paths will be developed for moving the new racks in the Fuel Building. The "new" racks will not be carried over any region of the pool containing fuel or safe shutdown equipment.

iv. The rack upending will be carried out in an area which is not poolside and will be qualified for a postulated rack drop from 6 feet elevation. Additionally, this area will not be overlapping to any safety related component.

v. All crew members involved in the reracking operation will be given training in the use of the lifting, upending equipment, and all other aspects of the reracking operation.

In addition to the above design, testing, and operation measures, the consequences of a postulated rack drop were also considered on the integrity of the pool structure. The following analysis was performed.

a. The heaviest rack module was postulated to free fall from the top of the water surface level to the pool floor.

b. The fall of a rack is assumed to occur in its normal vertical configuration which minimizes the retarding effect of water drag.

c. The falling rack is assumed to impact the pool slab undergoing an elastic/plastic impact.

The results of these calculations show that the maximum additional load on the pool structure is less than the capacity of the slab. Therefore, the integrity of the pool structure under the postulated rack drop event is ensured.

The fuel shuffle scheme developed for the spent fuel pool corresponding to the rack change-out presented in the preceding section is predicated on the following criteria:

1. No heavy load (rack or rig) with a potential to drop on a rack shall be carried over or near active fuel. This shall be accomplished by shuffling fuel into racks that are not in the area of the safe load path.
2. All heavy loads are lifted in such a manner that the C.G. of the lift point is aligned with the C.G. of the load being lifted.
3. Turnbuckles are utilized to "fine tune" the verticality of the rack being lifted.

All phases of the reracking activity will be conducted in accordance with written procedures which will be reviewed and approved in accordance with MP3 procedures.

The guidelines contained in NUREG-0612, Section 5 will be followed throughout the reracking activity. The guidelines of NUREG-0612 call for measures to "provide an adequate defense-in-depth for handling of heavy loads near spent fuel..." and cite four major causes of load handling accidents, namely

i. operator errors
ii. rigging failure
iii. lack of adequate inspection
iv. inadequate procedures

The MP3 rack expansion program ensures maximum emphasis on mitigation of the potential load drop accidents by implementing measures that will eliminate a possible accident during all aspects of the operation including the four aforementioned areas. A summary of the measures specifically planned to deal with the major causes is provided below.

Operator errors: As mentioned above, MP3 plans to provide comprehensive training to the installation crew.

Rigging failure: The lifting device designed for handling and installation of the racks in the MP3 fuel pool has redundancies in the lift legs, and lift eyes such that there are four independent load members. Failure of any one load bearing member would not lead to uncontrolled lowering of the load. The rig complies with all provisions of ANSI 14.6 - 1978, including compliance with the primary stress criteria, load testing at 300% of maximum lift load, and dye examination of critical welds.

The MP3 rig design is similar to the rigs used in the rerack of numerous other plants, such as Sequoyah, Zion, Salem, Three Mile Island Unit 1, D.C. Cook, and Connecticut Yankee.

Lack of adequate inspection: The designer of the racks will develop a set of inspection points which have proven to have eliminated any incidence of re-work or erroneous installation in numerous prior rerack projects. Inspection of lifting equipment will be performed per NUREG-0612.

Inadequate procedures: MP3 plans a multitude of procedures to cover the entire rerack effort, such as mobilization, rack handling, upending, lifting, installation, verticality, alignment, dummy gage testing, site safety, and ALARA compliance. Procedures for installation of new racks will be developed.

The series of operating procedures planned for MP3 rerack are the successors of the procedures implemented successfully in other projects.

In addition to the above, a complete inspection and preventive maintenance program of all the cranes and lifting equipment used in the project prior to the start of reracking are planned. Safe load paths will be developed as required by NUREG-0612.

Table 3.5 provides a synopsis of the requirements delineated in NUREG-0612, and our intended compliance.

In summary, the measures implemented in MP3 reracking are identical to those utilized in all recent reracks in the U.S., none of which has experienced any mishaps or reportable condition.

Table 3.1 BORAL EXPERIENCE LIST (Domestic and Foreign)			
PRESSURIZED WATER REACTORS			
Plant	**Utility**	**Vented Const-ruction**	**Mfg. Year**
Bellefonte 1, 2	Tennessee Valley Authority	No	1981
Donald C. Cook	Indiana & Michigan Electric	No	1979
Indian Point 3	NY Power Authority	Yes	1987
Maine Yankee	Maine Yankee Atomic Power	Yes	1977
Salem 1, 2	Public Service Electric & Gas	No	1980
Sequoyah 1,2	Tennessee Valley Authority	No	1979
Yankee Rowe	Yankee Atomic Power	Yes	1964/ 1983
Zion 1,2	Commonwealth Edison Company	Yes	1980
Byron 1,2	Commonwealth Edison Company	Yes	1988
Braidwood 1,2	Commonwealth Edison Company	Yes	1988
Yankee Rowe	Yankee Atomic Electric	Yes	1988
Three Mile Island	GPU Nuclear	Yes	1990
Sequoyah (rerack)	Tennessee Valley Authority	Yes	1992
Salem 1, 2	Public Service Electric & Gas	Yes	1994
Donald C. Cook (rerack)	American Electric Power	Yes	1992
BOILING WATER REACTORS			
Browns Ferry 1,2,3	Tennessee Valley Authority	Yes	1980
Brunswick 1.2	Carolina Power & Light	Yes	1981
Clinton	Illinois Power	Yes	1981

Table 3.1 **BORAL EXPERIENCE LIST (Domestic and Foreign)**			
Cooper	Nebraska Public Power	Yes	1979
Dresden 2,3	Commonwealth Edison Co.	Yes	1981
Duane Arnold	Iowa Electric Light and Power	No	1979
J.A. FitzPatrick	NY Power Authority	No	1978
E.I. Hatch 1,2	Georgia Power	Yes	1981
Hope Creek	Public Service Electric & Gas	Yes	1985
Humboldt Bay	Pacific Gas and Electric	Yes	1986
LaCrosse	Dairyland Power	Yes	1976
Limerick 1,2	PECO Nuclear	No	1980
Limerick 2	PECO Nuclear	Yes	1994
Monticello	Northern States Power	Yes	1978
Peach Bottom 2,3	PECO Nuclear	No	1980
Perry 1,2	Cleveland Elec. Illuminating	No	1979
Pilgrim	Boston Edison	No	1978
Susquehanna 1,2	Pennsylvania Power & Light	No	1979
Vermont Yankee	Vermont Yankee Atomic Power	Yes	1978/ 1986
Hope Creek	Public Service Electric & Gas	Yes	1989
Shearon Harris Pool B	Carolina Power & Light	Yes	1991
Duane Arnold	Iowa Electric Light & Power	Yes	1993
Pilgrim	Boston Edison Company	Yes	1993
LaSalle Unit 1	Commonwealth Edison Company	Yes	1992

Table 3.1 (continued)	
FOREIGN INSTALLATIONS USING BORAL	
England	
1 PWR Plant	Nuclear Electric plc.
France	
12 PWR Plants	Electricite de France
South Korea	
Ulchin 1.2	KEPCO
Kori 4	KEPCO
Yonggwang 1,2	KEPCO
South Africa	
Koeberg 1,2	ESCOM
Switzerland	
Beznau 1,2	Nordostschweizerische Kraftwerke AG
Gosgen	Kernkraftwerk Gosgen-Daniken AG
Taiwan	
Chinshan 1,2	Taiwan Power Company
Kuosheng 1,2	Taiwan Power Company
Mexico	
Laguna Verde Units 1,2	Comision Federal de Electricidad

Table 3.2

1100 ALLOY ALUMINUM PHYSICAL PROPERTIES

Density	0.098 lb/cu. in. 2.713 gm/cc
Melting Range	1190-1215 deg. F 643-657 deg. C
Thermal Conductivity (77 deg. F)	128 Btu/hr/sq ft/deg. F/ft 0.53 cal/sec/sq cm/deg. C/cm
Coefficient of Thermal Expansion (68-212 deg. F)	13.1 x 10^{-6} in/in., °F 23.6 x 10^{-6} cm/cm, °C
Specific heat (221 deg. F)	0.22 Btu/lb/deg. F 0.23 cal/gm/deg. C
Modulus of Elasticity	10×10^6 psi
Tensile Strength (75 deg. F)	13,000 psi annealed 18,000 psi as rolled
Yield Strength (75 deg. F)	5,000 psi annealed 17,000 psi as rolled
Elongation (75 deg. F)	35-45% annealed 9-20% as rolled
Hardness (Brinell)	23 annealed 32 as rolled
Annealing Temperature	650 deg. F 343 deg. C

Table 3.3 **CHEMICAL COMPOSITION - 1100 ALLOY ALUMINUM**	
99.00% min.	Aluminum
1.00% max.	Silicone and Iron
0.05-0.20% max.	Copper
0.05% max.	Manganese
0.10% max.	Zinc
0.15% max.	others each

Table 3.4	
BORON CARBIDE CHEMICAL COMPOSITION, WEIGHT %	
Total boron	70.0 min.
B^{10} isotopic content in natural boron	18.0
Boric oxide	3.0 max.
Iron	2.0 max.
Total boron plus total carbon	94.0 min.
BORON CARBIDE PHYSICAL PROPERTIES	
Chemical formula	B_4C
Boron content (weight)	78.28%
Carbon content (weight)	21.72%
Crystal Structure	rombohedral
Density	2.51 gm/cc 0.0907 lb/cu.in.
Melting Point	2450°C-4442°F
Boiling Point	3500°C-6332°F
Microscopic Capture cross-section	600 barn

Table 3.5

HEAVY LOAD HANDLING COMPLIANCE MATRIX (NUREG-0612)

	Criterion	Compliance
1.	Are safe load paths defined for the movement of heavy loads to minimize the potential of impact, if dropped on irradiated fuel and safe shutdown equipment?	Yes
2.	Will procedures be developed to cover: identification of required equipment, inspection, and acceptance criteria required before movement of load, steps and proper sequence for handling the load, defining the safe load paths, and special precautions?	Yes
3.	Will crane operators be trained and qualified?	Yes
4.	Will special lifting devices meet the guidelines of ANSI 14.6-1978?	Yes
5.	Will non-customer lifting devices be installed and used in accordance with ANSI B30.9-1971?	Yes
6.	Will the cranes be inspected and tested prior to use in rerack?	Yes
7.	Does the crane meet the intent of ANSI B30.2-1976 and CMMA-70?	Yes

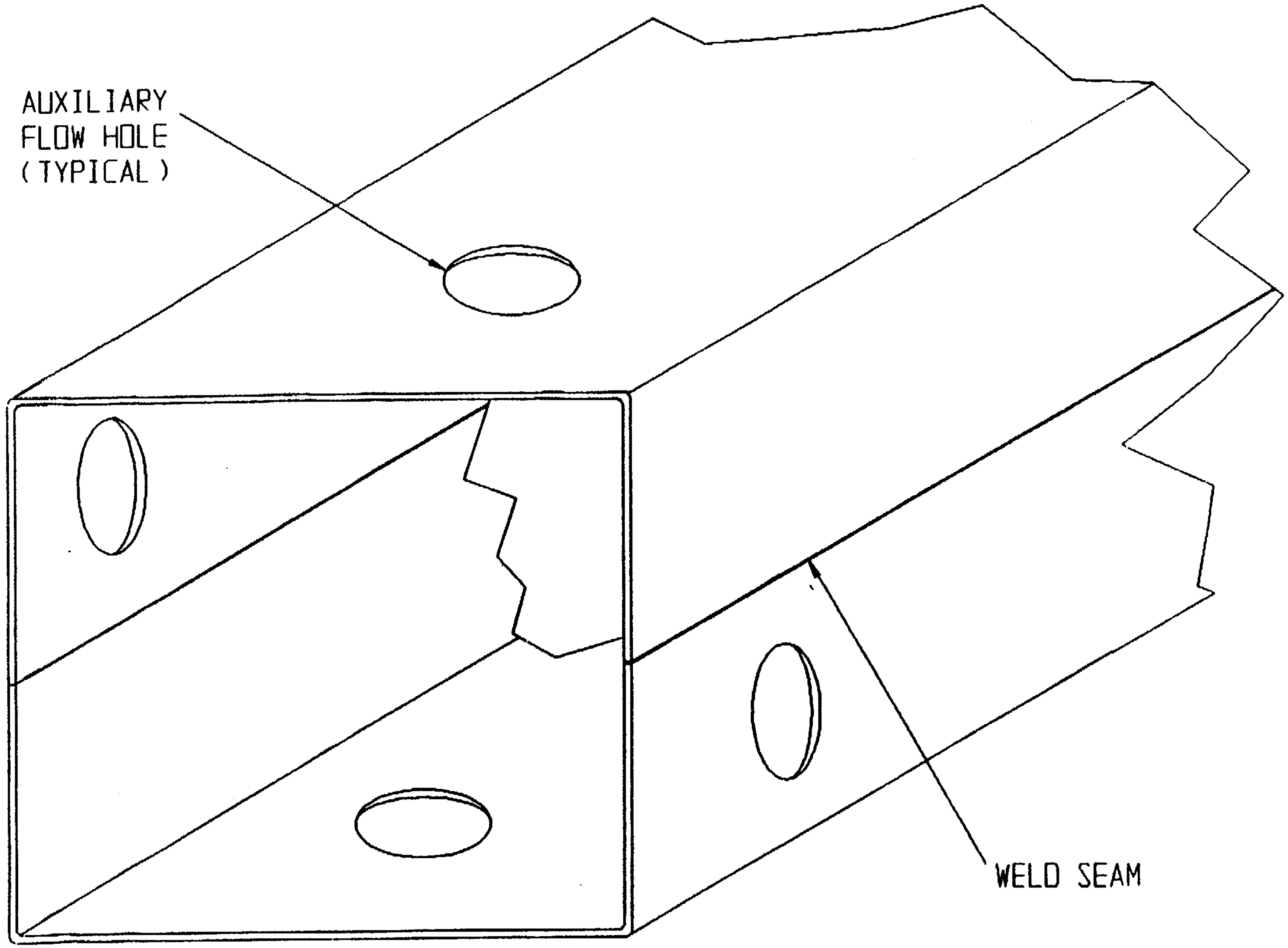


FIGURE 3.1; SEAM WELDED PRECISION FORMED CHANNELS

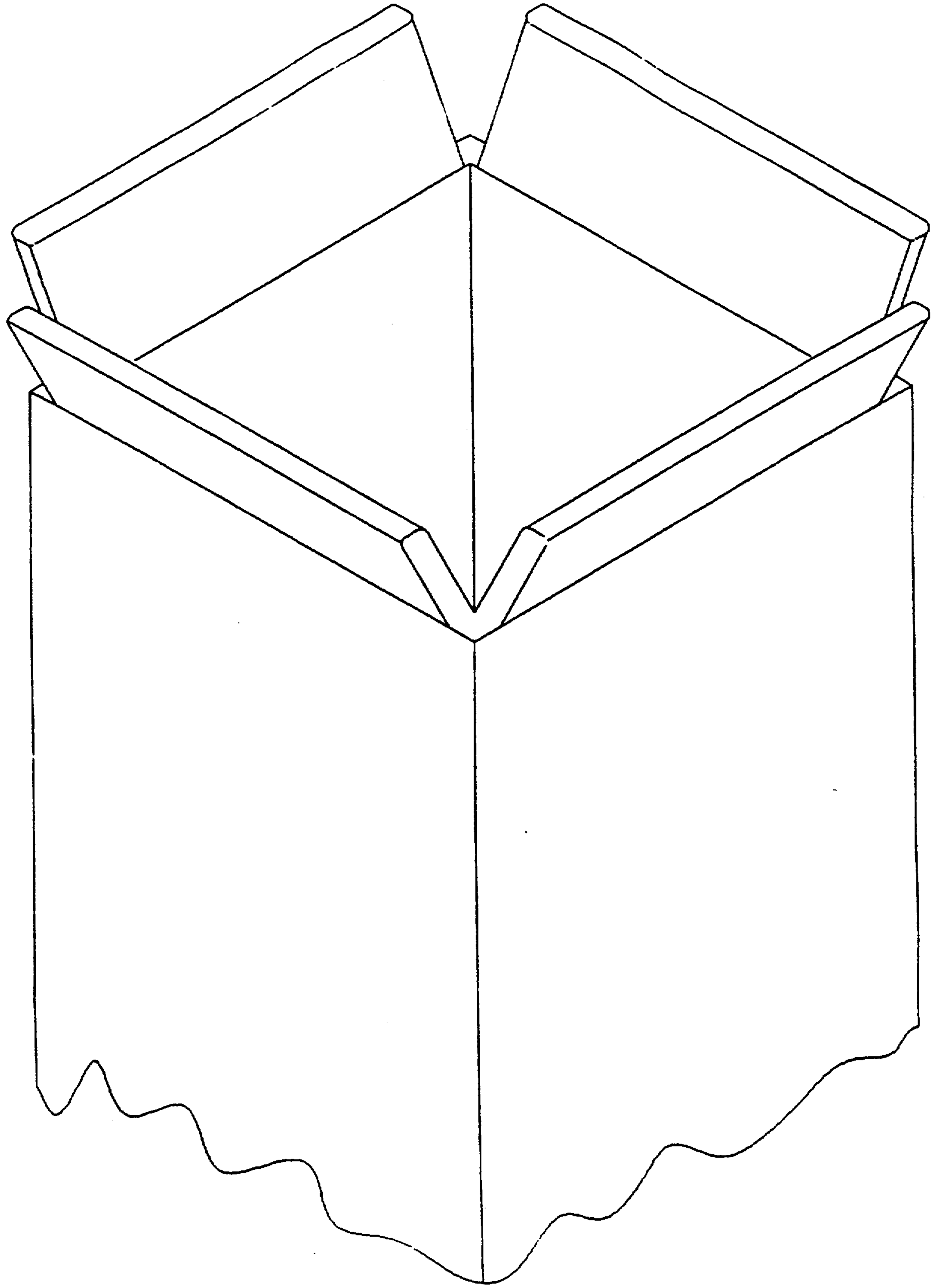


FIGURE 3.2: TAPERED REGION 1 CELL END

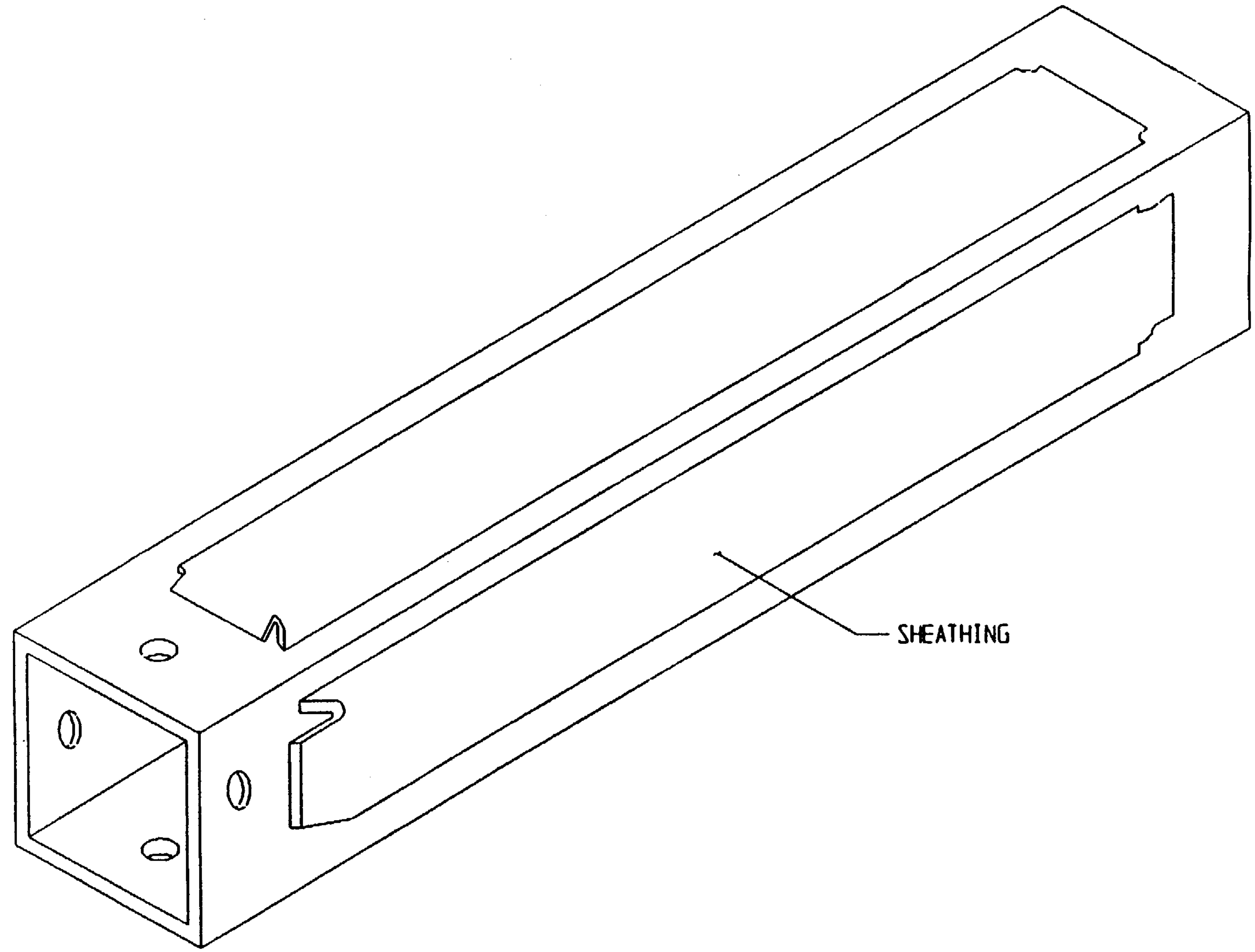


FIGURE 3.3 COMPOSITE BOX ASSEMBLY

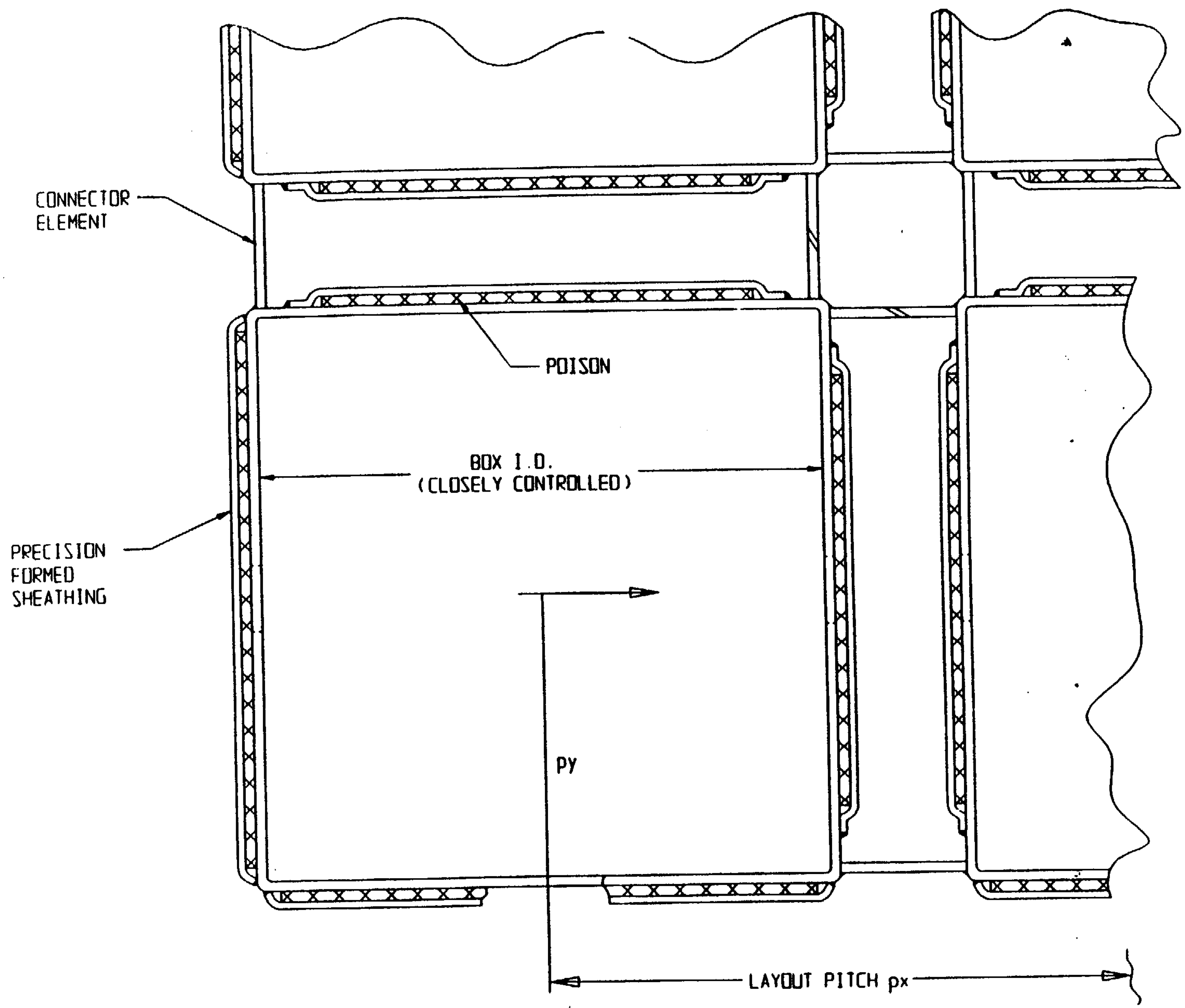


FIGURE 3.4: ASSEMBLAGE OF REGION 1 CELLS

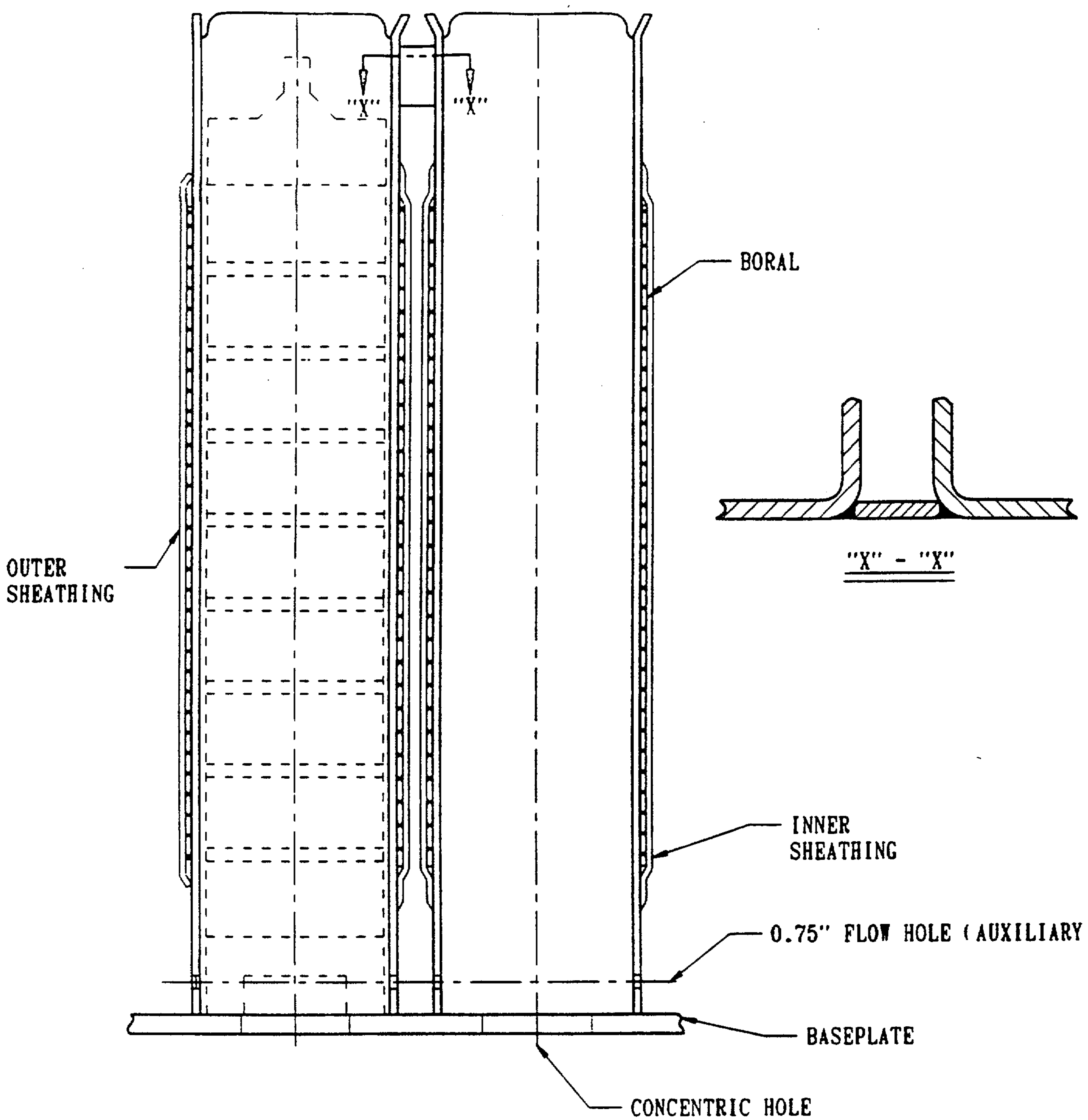


FIGURE 3.5; ELEVATION VIEW OF REGION 1 RACK

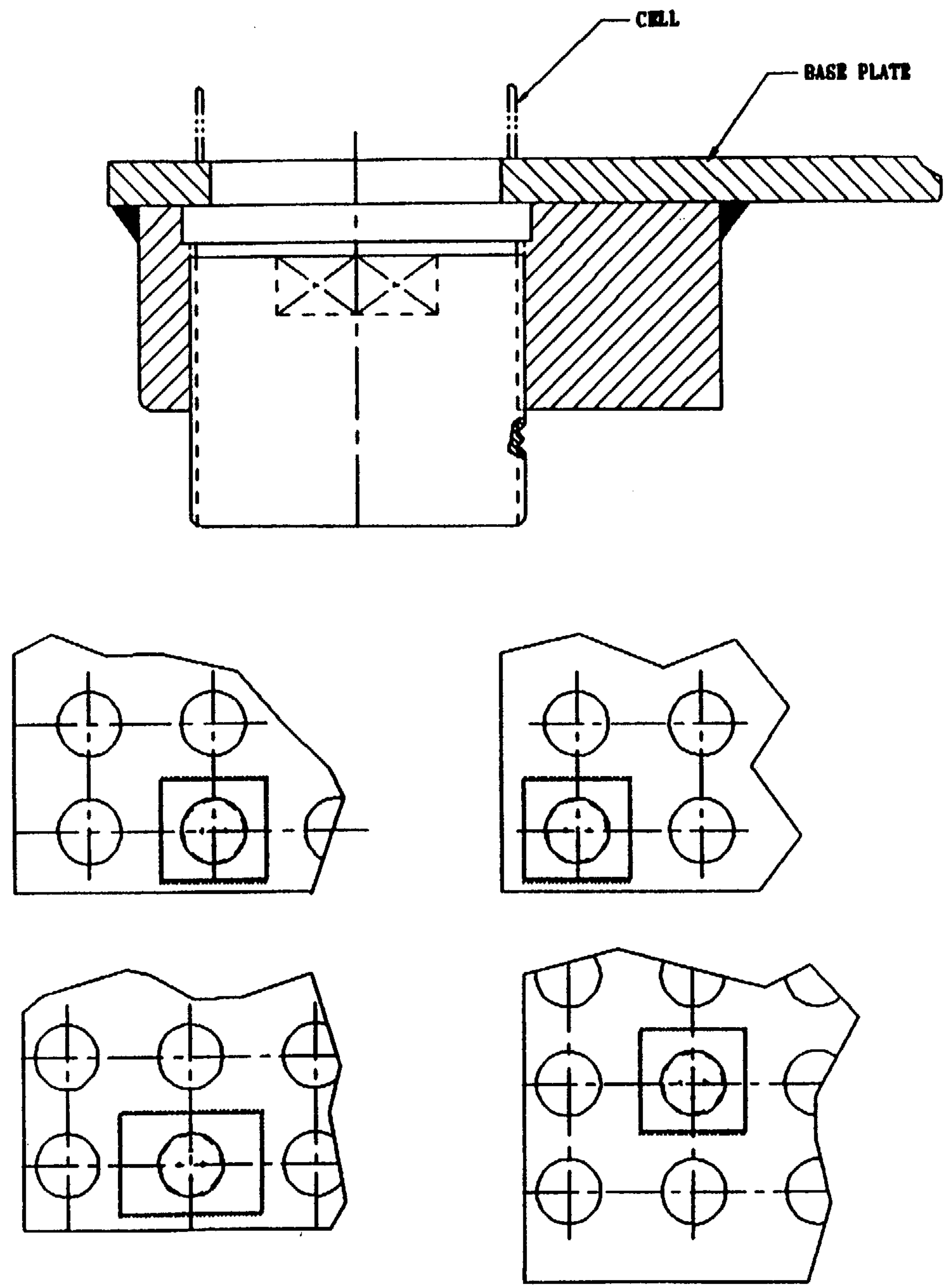


FIGURE 3.6; SUPPORT PEDESTAL FOR HOLTEC RACKS

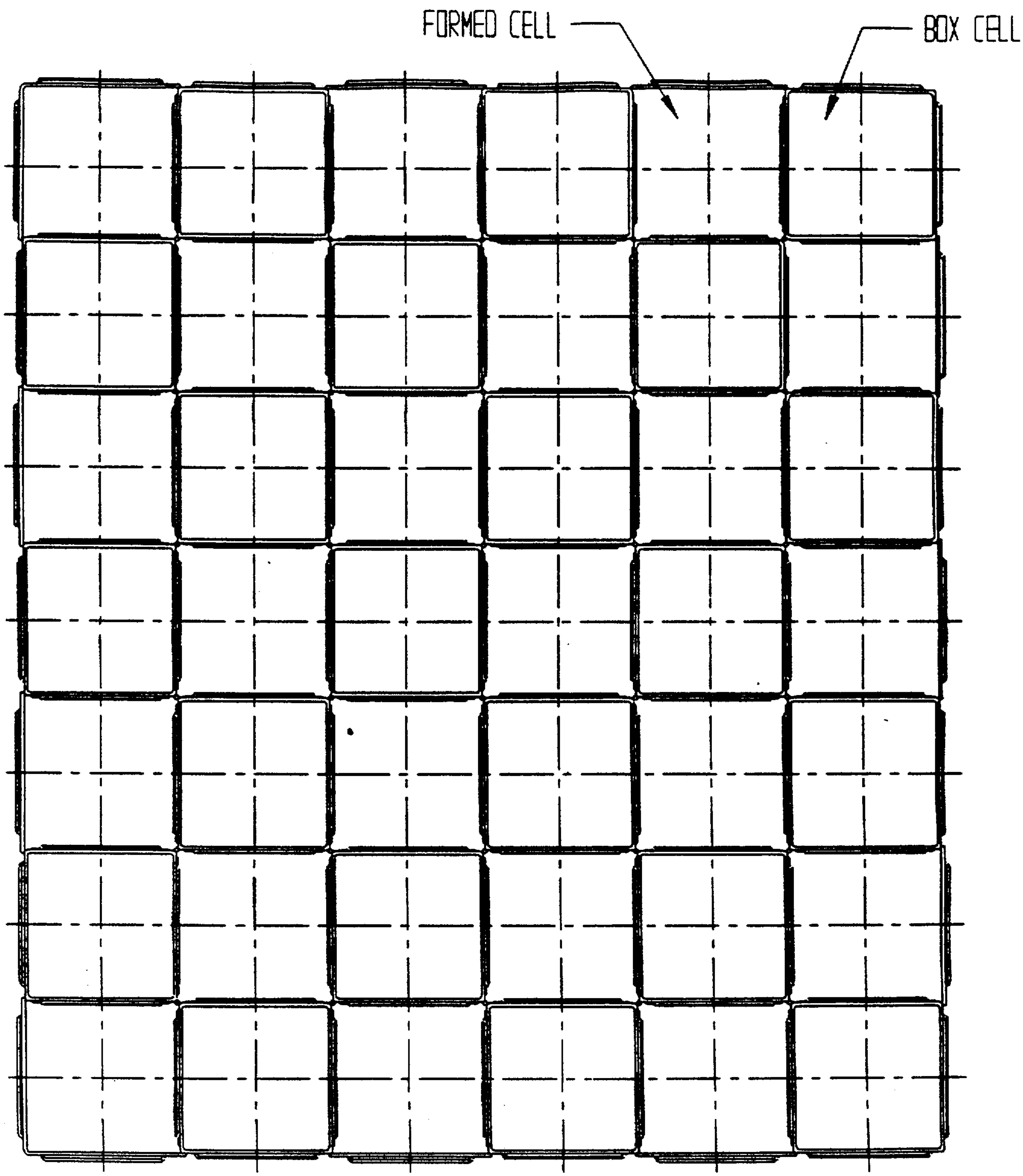


FIGURE 3.7; TYPICAL ARRAY OF REGION 2 CELLS
(NON-FLUX TRAP CONSTRUCTION)

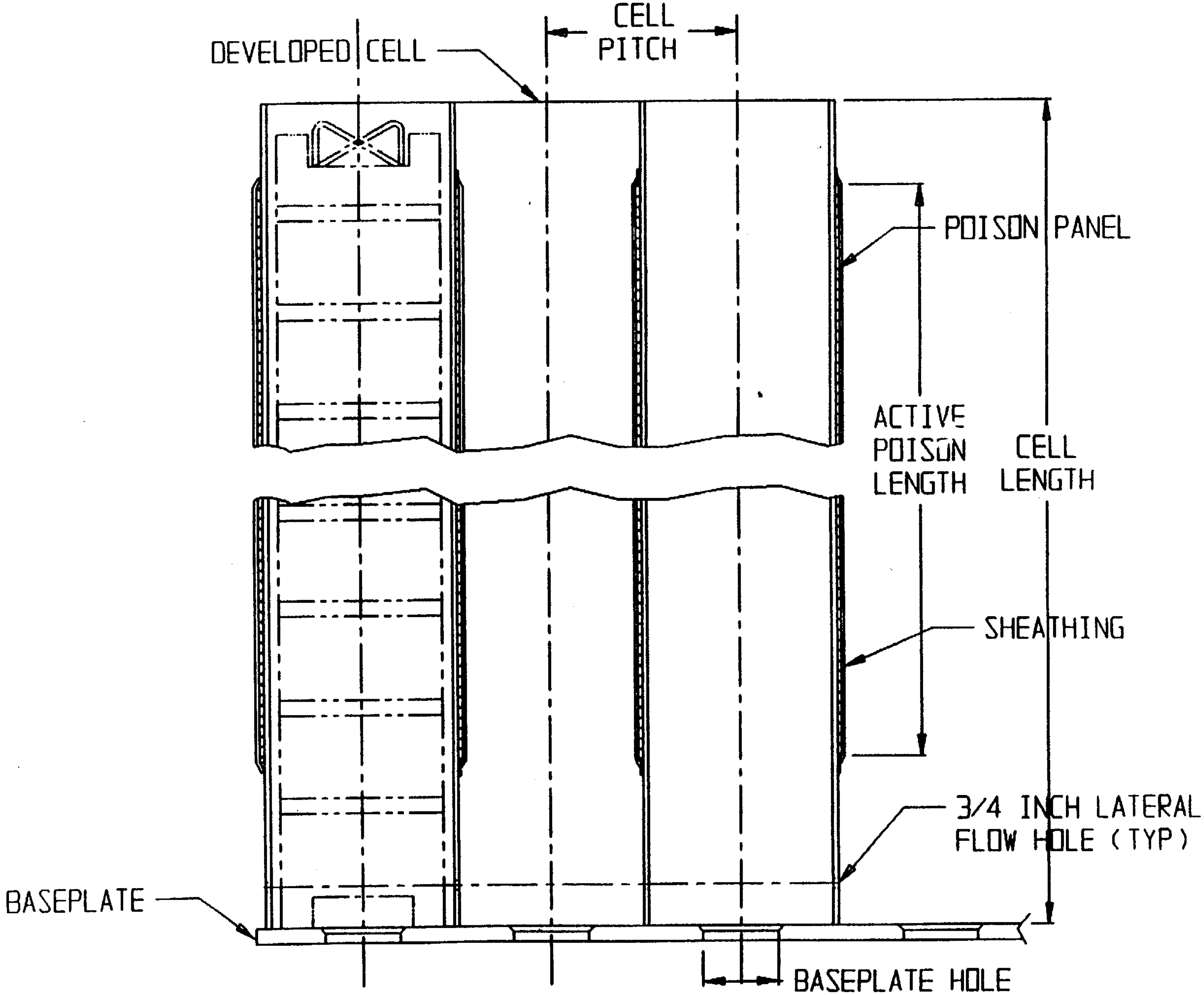


FIGURE 3.8; ELEVATION VIEW OF REGION 2 RACK MODULE

4.0 CRITICALITY SAFETY EVALUATION

4.1 DESIGN BASES

The high density spent fuel storage racks for Millstone Unit 3 are designed to assure that the effective neutron multiplication factor, k_{eff}, is equal to or less than 0.945 with the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with un-borated water at a temperature within the operating range corresponding to the highest reactivity. Including all applicable uncertainties, the maximum k_{eff} is shown to be less than or equal to 0.945 with a 95% probability at a 95% confidence level [4.1.1]. Reactivity effects of abnormal and accident conditions have also been evaluated to assure that under credible abnormal and accident conditions, the reactivity will not exceed 0.945.

Applicable codes, standards, and regulations or pertinent sections thereof, include the following:

- *Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3 - July 1981.
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.
- L.I. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," June 1998.
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December 1981.
- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.

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

USNRC guidelines and the applicable ANSI standards specify that the maximum effective multiplication factor, k_{eff}, including bias, uncertainties, and calculational statistics, shall be less than or equal to 0.95, with 95% probability at the 95% confidence level. In the present criticality safety evaluation, the design limit was assumed to be 0.945, which is more conservative than the limit specified in the regulatory guidelines.

To ensure that the true reactivity will always be less than the calculated reactivity, the following conservative assumptions were made:

- Moderator is un-borated water at a temperature within the operating range that results in the highest reactivity.
- The racks were assumed to be fully loaded with the most reactive fuel authorized to be stored in the racks without any control rods or burnable poison, such as Integral Fuel Burnable Absorber (IFBA) rods.
- No soluble poison (boron) is assumed to be present in the pool water under normal operating conditions.
- Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water.
- The effective multiplication factor of an infinite radial array of fuel assemblies was used except for the assessment of peripheral effects and certain abnormal/accident conditions where neutron leakage is inherent.
- In-core depletion calculations assume conservative operating conditions, highest fuel and moderator temperature, and an allowance for the soluble boron concentrations during in-core operations.
- All Region 3 analyses assume that the Boraflex is replaced by water, and thus, no credit is taken for neutron absorption in the Boraflex panels.

The spent fuel storage racks are designed to accommodate the fuel assembly types listed in Table

4.1.1 with a maximum enrichment of 5 wt% ^{235}U. Although the two assembly types listed in Table 4.1.1 are nearly identical, differing only in the guide and instrument tube dimensions, the Westinghouse 17x17 Vantage 5H (V5H) assembly was determined to have very slightly higher reactivity. Therefore, the V5H assembly was used as the design basis fuel assembly.

Three separate storage regions are provided in the spent fuel pool. The independent acceptance criteria for storage in each of the regions are as follows:

⇒ Region 1 is designed to accommodate (1) new un-irradiated fuel assemblies with a maximum nominal enrichment of 5.0 wt% ^{235}U in a 3-out-of-4 arrangement with the fourth cell empty and blocked and (2) fuel assemblies in a 4-out-of-4 arrangement (unrestricted) with a maximum nominal enrichment of 5.0 wt% ^{235}U which have accumulated a minimum burnup of 8.0 MWd/kgU or fuel of initial enrichment and burnup combinations within the acceptable domain depicted in Figure 4.1.1.

⇒ Region 2 is designed to accommodate fuel assemblies with a maximum nominal enrichment of 5.0 wt% ^{235}U which have accumulated a minimum burnup of 39.0 MWd/kgU or fuel of initial enrichment and burnup combinations within the acceptable domain depicted in Figure 4.1.2.

⇒ Region 3 is designed to accommodate fuel assemblies with a maximum nominal enrichment of 5.0 wt% ^{235}U which have accumulated minimum burnup and cooling times that fall within the acceptable domains depicted in Figure 4.1.3.

The water in the spent fuel storage pool normally contains soluble boron which would result in a large sub-criticality margin under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 for normal storage be evaluated for the accident condition that assumes the loss of soluble boron. The double contingency principle of ANSI N-16.1-1975 and of the April 1978 NRC letter allows credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time. Consequences of abnormal and accident conditions have also been evaluated, where "abnormal" refers to conditions which may reasonably be expected to occur during the lifetime of the plant and "accident" refers to conditions which are not expected to occur but nevertheless must be protected against.

4.2 SUMMARY OF CRITICALITY ANALYSES

4.2.1 Normal Operating Conditions

The criticality analyses for each of the three separate regions of the spent fuel storage pool are summarized in Tables 4.2.1, 4.2.3, and 4.2.5, for the design basis storage conditions. For the acceptance criteria defined in the previous section, the maximum k_{eff} values are shown to be less than or equal to 0.945 (95% probability at the 95% confidence level) in each of the three regions.

4.2.1.1 Region 1

Calculations have been performed to qualify the Region 1 racks for storage of new un-irradiated fuel assemblies with a maximum nominal enrichment of 5.0 wt% ^{235}U in a 3-out-of-4 arrangement with the fourth cell empty and blocked and in a 4-out-of-4 arrangement (unrestricted) with initial enrichment and burnup combinations within the acceptable domain depicted in Figure 4.1.1. The criticality analyses for Region 1 of the spent fuel storage pool are summarized in Table 4.2.1, and demonstrate that for the defined acceptance criteria, the maximum k_{eff} is less than 0.93.

The data points shown in Figure 4.1.1 are tabulated in Table 4.2.2. For convenience, the minimum (limiting) burnup data may be described as a function of the nominal initial enrichment, *E*, in wt% ^{235}U by a bounding polynomial expression as follows:

$$B = -0.6667 \times E^2 + 12.093 \times E - 35.798.$$

where *B* is the minimum burnup in MWd/kgU and *E* is the enrichment in wt% ^{235}U (for initial enrichments up to 5.0 wt% ^{235}U). Alternatively, because the data are nearly linear, linear interpolation between the points listed in Table 4.2.2 is also acceptable.

4.2.1.1.1 Interface Between Storage Arrangements

The two different storage arrangements that are available in the Region 1 racks (i.e., 3-out-of-4 and 4-out-of-4) may be utilized in any of the Region 1 racks, including both arrangements in a single rack, provided the following interface requirement is met; the row in the 3-out-of-4 storage area bordering the interface between adjacent 3-out-of-4 and 4-out-of-4 storage areas must contain alternating cell blockers. The interface requirement is illustrated in Figure 4.2.1. A calculation was performed to demonstrate that such an arrangement is less reactive than either of the individual arrangements alone.

During the rack installation, cell blocking devices will be installed in a manner consistent with the aforementioned requirement. The interface requirement will be ensured through administrative procedures. A cell blocking device may be removed, provided all adjacent and diagonal fuel assemblies around the cell blocking device are removed beforehand.

4.2.1.2 Region 2

Calculations have been performed to qualify the Region 2 racks for storage of fuel assemblies with a maximum nominal initial enrichment of 5.0 wt% ^{235}U which have accumulated a minimum burnup of 39.0 MWd/kgU or fuel of initial enrichment and burnup combinations within the acceptable domain depicted in Figure 4.1.2. The criticality analyses for Region 2 of the spent fuel storage pool are summarized in Table 4.2.3, and demonstrate that for the defined acceptance criteria, the maximum k_{eff} is less than 0.945.

The calculated maximum reactivity in Region 2 includes the reactivity effect of the axial distribution in burnup and provides an additional margin of uncertainty for the depletion calculations. The data points shown in Figure 4.1.2 are tabulated in Table 4.2.4. For convenience, the minimum (limiting) burnup data may be described as a function of the nominal initial enrichment, *E*, in wt% ^{235}U by a bounding polynomial expression as follows:

$$B = -0.4608 \times E^4 + 6.641 \times E^3 - 34.854 \times E^2 + 90.385 \times E - 83.40.$$

where *B* is the minimum burnup in MWd/kgU and *E* is the enrichment in wt% ^{235}U (for initial enrichments from 2.0 to 5.0 wt% ^{235}U). Fuel assemblies with enrichments less than 2.0 wt% ^{235}U will conservatively be required to meet the burnup requirements of 2.0 wt% ^{235}U assemblies as shown in Fig 4.1.2. Alternatively, because the data are nearly linear, linear interpolation between the points listed in Table 4.2.4 is also acceptable.

4.2.1.3 Region 3

Calculations have been performed to qualify the existing Westinghouse designed racks, referred to herein as Region 3 racks, for storage of fuel assemblies with a maximum nominal initial enrichment of 5.0 wt% ^{235}U which have accumulated minimum burnup and cooling times that fall within the acceptable domains depicted Figure 4.1.3. The criticality analyses for Region 3 of the spent fuel storage pool are summarized in Table 4.2.5 and demonstrate that the maximum k_{eff} is equal to 0.945. which conforms to the defined acceptance criterion.

The calculated maximum reactivity in Region 3 includes the reactivity effect of the axial distribution in burnup and provides an additional margin of uncertainty for the depletion calculations. The data points shown in Figure 4.1.3 are tabulated in Table 4.2.6. For convenience, the minimum (limiting) burnup data for each of the cooling times shown in Figure 4.1.3 may be described as a function of the nominal initial enrichment, *E*, in wt% ^{235}U by bounding polynomial expressions as follows:

Cooling Time (years)	Polynomial Expression

0	$B = 0.1000 \times E^2 + 14.1696 \times E - 17.5390$
5	$B = 0.4651 \times E^2 + 10.0120 \times E - 11.3919$
10	$B = 0.6730 \times E^2 + 7.8408 \times E - 8.3853$
20	$B = 0.6151 \times E^2 + 7.3547 \times E - 7.9121$

where B is the minimum burnup in MWd/kgU and E is the enrichment in wt% ^{235}U (for initial enrichments from 2.0 to 5.0 wt% ^{235}U). Fuel assemblies with enrichments less than 2.0 wt% ^{235}U will conservatively be required to meet the burnup requirements of 2.0 wt% ^{235}U assemblies as shown in Fig 4.1.3. Alternatively, because the data are nearly linear, linear interpolation between the points listed in Table 4.2.6 is also acceptable.

The burnup criteria identified above for acceptable storage in each of the three regions will be implemented by appropriate administrative procedures to ensure verified burnup as specified in the proposed Regulatory Guide 1.13, Revision 2.

4.2.2 Abnormal and Accident Conditions

Although credit for the soluble poison normally present in the spent fuel pool water is permitted under abnormal or accident conditions, most abnormal or accident conditions will not result in exceeding the limiting reactivity even in the absence of soluble poison. The effects on reactivity of credible abnormal and accident conditions are discussed in Section 4.8 and summarized in Tables 4.2.7 and 4.2.8. Strict administrative procedures to assure the presence of soluble poison will preclude the possibility of the simultaneous occurrence of the two independent accident conditions.

The inadvertent misplacement of a fresh fuel assembly has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. Assuring the presence of soluble poison during fuel handling operations will preclude the possibility of the simultaneous occurrence of the two independent accident conditions. The largest reactivity increase would occur if a fresh fuel assembly of 5.0 wt% ^{235}U enrichment were to be inadvertently loaded into an empty cell in Region 3 with the remainder of the rack fully loaded with fuel of the highest permissible reactivity. Under this accident condition, credit for the presence of soluble poison is permitted by the NRC guidelines. Calculations indicate that 800 ppm soluble boron, that is to be required by the Technical Specifications during fuel handling operations, is more than adequate to assure that the limiting k_{eff} of 0.945 is not exceeded.

With the assumption that the Boraflex panels are replaced by water, the moderator temperature coefficient of reactivity in Region 3 is positive. Therefore, an increase in the spent fuel pool temperature above the normal operating conditions (i.e., above 160 F), has the potential for exceeding the limiting reactivity in Region 3, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. The largest reactivity increase would occur if boiling took place in Region 3 with the remainder of the rack fully loaded with fuel of the highest permissible reactivity. Calculations indicate that 100 ppm soluble boron is more than adequate to assure that the limiting k_{eff} of 0.945 is not exceeded for temperatures greater than 160 F and boiling.

However, since the spent fuel pool cooling system is capable of maintaining fuel pool water temperature less than 160 F even with a single failure, this calculation is outside of the design basis, and no further action is necessary.

† Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Reg. Guide 1.13 (Section 1.4, Appendix A).

4.3 REFERENCE FUEL STORAGE CELLS

4.3.1 Reference Fuel Assembly

The design basis fuel assembly is the Westinghouse 17x17 Vantage 5H (V5H) assembly. Table 4.1.1 summarizes the fuel assembly design specifications.

4.3.2 Region 1 Fuel Storage Cells

Figure 4.3.1 shows the calculational model of the nominal Region 1 spent fuel storage cell containing a 17x17 V5H assembly. The Region 1 storage cells are composed of stainless steel boxes separated by a gap with fixed neutron absorber panels, Boral, on each of the box walls. The [redacted] hick steel walls define the storage cells which have a [redacted] nch nominal inside dimension. A [redacted] inch stainless steel sheath supports the Boral panel and defines the boundary of the flux-trap water-gap used to augment reactivity control. The cells are located on a lattice spacing of [redacted] nch in one direction and [redacted] nch in the other direction. Stainless steel channels connect the storage cells in a rigid structure and define the flux-trap between the Boral panels, which are [redacted] inch in one direction and [redacted] inch in the other direction. The Boral absorber has a thickness of [redacted] nch and a nominal B-10 areal density of [redacted] The Boral absorber panels are [redacted] inches in width and [redacted] inches in length. Boral panels are installed on all exterior walls facing other racks, as well as, non-fueled regions, i.e., the pool walls. The minimum gap between neighboring Region 1 style racks and between Region 1 and Region 2 style racks is 1.5 inches. Region 1 and Region 3 racks are not located adjacent to one another.

4.3.3 Region 2 Fuel Storage Cells

Figure 4.3.2 shows the calculational model of the nominal Region 2 spent fuel storage cell containing a 17x17 V5H assembly. The Region 2 storage cells are composed of stainless steel walls with a single fixed neutron absorber panel, Boral, (attached by a . inch stainless steel sheathing) centered on each side in [redacted] inch channel. Stainless steel boxes are arranged in an alternating pattern such that the connection of the box corners form storage cells between those of the stainless steel boxes. These cells are located on a lattice spacing o[redacted] inch. The [redacted]hick steel walls define a storage cell which has [redacted]ch nominal inside dimension. The Boral absorber has a thickness o[redacted]nch and a nominal B-10 areal density of [redacted] The Boral absorber panels are [redacted] inches in width and [redacted] inches in length. Boral panels are installed on all exterior walls facing other racks, as well as, non-fueled regions, i.e., the pool walls. The minimum gap between neighboring Region 2 style racks is 0.50 inches, while the minimum gap between Region 1 and Region 2 style racks is 1.5 inches. The minimum gap between Region 2 and Region 3 racks is 1.28 inches.

4.3.4 Region 3 Fuel Storage Cells

Figure 4.3.3 shows the calculational model of the nominal Region 3 spent fuel storage cell containing a 17x17 V5H assembly. The Region 3 storage cells are composed of stainless steel boxes separated by a gap with fixed neutron absorber panels, Boraflex, on each of the box walls. The [redacted] thick steel walls define the storage cells which have [redacted]nch nominal inside dimension. A [redacted] inch stainless steel sheath supports the Boraflex panel and defines the boundary of the flux-trap water-gap used to augment reactivity control. The cells are located on a lattice spacing of [redacted] The Boraflex absorber has a thickness o[redacted]nch and a nominal B-10 areal density of approximately [redacted] The Boraflex absorber panels are [redacted] inches in width. However, all Region 3 analyses assume that the Boraflex is replaced by water, and thus, no credit is taken for neutron absorption in the Boraflex panels. The minimum gap between Region 3 and Region 2 style racks is 1.28 inches and the minimum gap between Region 3 and Region 1 style racks is 76.09 inches. Region 3 and Region 1 racks are not located adjacent to one another.

4.4 ANALYTICAL METHODOLOGY

4.4.1 Reference Design Calculations

The principal methods for the criticality analyses of the high density storage racks include the following codes: (1) MCNP4a [4.4.1], (2) KENO5a [4.4.2], and CASMO-3 [4.4.5-4.4.7]. MCNP4a is a continuous energy three-dimensional Monte Carlo code developed at the Los Alamos National Laboratory. KENO5a is a three-dimensional multigroup Monte Carlo code developed at the Oak Ridge National Laboratory as part of the SCALE 4.3 package [4.4.3]. The KENO5a calculations used the 238-group SCALE cross-section library and NITAWL [4.4.4] for ^{238}U resonance shielding effects (Nordheim integral treatment). Benchmark calculations, presented in Appendix 4A, indicate a bias of 0.0009 with an uncertainty of 0.0011 for MCNP4a and 0.0030 0.0012 for KENO5a, both evaluated with the 95% probability at the 95% confidence level [4.1.1].

Fuel depletion analyses during core operation were performed with CASMO-3, a two-dimensional multigroup transport theory code based on capture probabilities [4.4.5 - 4.4.7]. Restarting the CASMO-3 calculations in the storage rack geometry yields the two-dimensional infinite multiplication factor (k) for the storage rack. Parallel calculations with CASMO-3 for the storage rack at various enrichments enable a reactivity equivalent enrichment (fresh fuel) to be determined that provides the same reactivity in the rack as the depleted fuel. CASMO-3 was also used to determine the small reactivity uncertainties (differential calculations) of manufacturing tolerances and the reactivity effect of various decay times (for Region 3 only).

In the geometric models used for the calculations, each fuel rod and its cladding were described explicitly and reflecting boundary conditions were used in the radial direction which has the effect of creating an infinite radial array of storage cells. Monte Carlo calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the MCNP4a and KENO5a calculated reactivities and to assure convergence, a minimum of 1 million neutron histories were accumulated in each calculation.

4.4.2 Fuel Burnup Calculations and Uncertainties

CASMO-3 was used for burnup calculations in the hot operating condition. CASMO-3 has been extensively benchmarked [4.4.7, 4.4.8] against cold, clean, critical experiments (including plutonium-bearing fuel), Monte Carlo calculations, reactor operations, and heavy element concentrations in irradiated fuel. In addition to burnup calculations, CASMO-3 was used for evaluating the small reactivity increments (by differential calculations) associated with manufacturing tolerances, for determining temperature effects, and the reactivity effects of decay time.

In the CASMO-3 geometric models, each fuel rod and its cladding were described explicitly and reflective boundary conditions were used between storage cells. These boundary conditions have the effect of creating an infinite array of storage cells.

Conservative assumptions of moderator and fuel temperatures and the average operating soluble boron concentrations were used to assure the highest plutonium production and hence conservatively high values of reactivity during burnup. Since critical experiment data with spent fuel is not available for determining the uncertainty in depletion calculations, an allowance for uncertainty in reactivity was assigned based upon other considerations. Assuming the uncertainty in depletion calculations is less than 5% of the total reactivity decrement, a burnup dependent uncertainty in reactivity for burnup calculations was assigned. Thus, the burnup uncertainty varies (increases) with burnup. This allowance for burnup uncertainty was included in determination of the acceptable burnup versus enrichment combinations, and is believed to be a conservative estimate.

† The majority of the uncertainty in depletion calculations derives from uncertainties in fuel and moderator temperatures and the effect of reactivity control methods (e.g., soluble boron). For depletion calculations, bounding values of these operating parameters were assumed to assure conservative results in the analyses.

4.4.3 Effect of Axial Burnup Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower regions. At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of high neutron leakage. Consequently, it is expected that over most of the burnup history, fuel assemblies with distributed burnups will exhibit a slightly lower reactivity than that calculated for the uniform average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup.

Among others, Turner [4.4.9] has provided generic analytic results of the axial burnup effect based upon calculated and measured axial burnup distributions. These analyses confirm the minor and generally negative reactivity effect of the axially distributed burnups at values less than about 27 MWd/kgU with small positive reactivity effects at higher burnup values. Because of the decay of ^{241}Pu, the effect of the axial burnup distribution becomes larger when cooling times are considered. For the present criticality analyses, the reference calculations utilized representative axial burnup distributions previously calculated for Millstone Unit 3. Burnup-equivalent enrichments were determined with CASMO-3 for each of 24 axial zones and used in three-dimensional Monte Carlo calculations. Results of these calculations, therefore, inherently include the effect of the axial distribution in burnup. Comparison of these results to results of calculations with uniform axial burnup allows the reactivity effect of the axial burnup distribution to be quantified. This reactivity effect is included, where applicable, in the calculation of the maximum k_{eff} values. For Region 3, where credit for cooling time is considered, calculations were performed to determine the reactivity effect at each of the cooling times.

4.4.4 Long-Term Changes in Reactivity

Since the fuel racks in Region 3 are intended to contain spent fuel for long periods of time, consideration was given to the long-term changes in reactivity of spent fuel. Calculations confirm that reactivity continuously decreases as the spent fuel ages. Early in the decay period, Xenon grows from Iodine decay (reducing reactivity) and subsequently decays, with the reactivity reaching a maximum at about 100 hours. To assure conservatism in the restart calculations, the Xe-135 is set to zero. The decay of Pu-241 (13-year half-life) and growth of Am-241 substantially reduce the reactivity during long term storage. Figure 4.1.3 illustrates the reduction in reactivity during long term storage. For Region 3 racks, credit is taken for this long-term reduction in reactivity, and includes the increased effect of the axial burnup distribution. However, for Regions 1 and 2, no credit is taken for this long-term reduction in reactivity, other than to indicate an increasing subcriticality margin.

4.5 REGION 1 CRITICALITY ANALYSES AND TOLERANCES

4.5.1 Nominal Design Case

For the nominal storage cell design in Region 1, the criticality safety analyses are summarized in Table 4.2.1. These data confirm that the maximum reactivity in Region 1 remains conservatively less than the regulatory limit (k_{eff} 0.95). An independent calculation with the KENO5a code provides confirmation of the validity of the reference MCNP4a calculations.

4.5.2 Uncertainties Due to Burnup

For storage in the 3-out-of-4 arrangement, consideration of fuel burnup is not necessary, and thus, burnup related uncertainties are not applicable. However, for unrestricted storage in the 4-out-of-4 arrangement, fuel burnup is required. CASMO-3 was used for the depletion analysis and the restart option was used to analytically transfer the spent fuel into the storage rack configuration at a reference temperature of 4 C (corresponding to the highest reactivity, see Section 4.8.1). Calculations were also made for fuel of several different initial enrichments and interpolated to define the burnup-dependent equivalent enrichments , at each burnup. MCNP4a calculations were then made for the equivalent enrichment to establish the limiting k_{eff} value, which includes all applicable uncertainties. At the limiting burnups required for Region 1 storage, the effect of the axial distribution in burnup is negative, and thus, is not included. These calculations were used to define the boundary of the acceptable domain shown in Figure 4.1.1.

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† The (reactivity) equivalent enrichment is the fresh un-burned fuel enrichment that yields the same reactivity as the depleted fuel, both evaluated in the storage rack configuration. The equivalent enrichment may then be used in three-dimensional MCNP4a or KENO5a calculations.

4.5.3 Uncertainties Due to Tolerances

The reactivity effects of manufacturing tolerances are tabulated, along with the tolerances, in Table 4.5.1. The individual tolerances were conservatively calculated for the design basis fresh unburned fuel assembly.

4.5.4 Eccentric Fuel Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell. However, calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). These calculations indicated that the reactivity effect is small and negative. Therefore, the reference case in which the fuel assemblies are centered is controlling and no uncertainty for eccentricity is necessary.

4.5.5 Water-Gap Spacing Between Racks

The minimum water-gap between racks, which is 1.5 inches between neighboring Region 1 style racks and also 1.5 inches between Region 1 and Region 2 style racks, constitutes a neutron flux-trap for the storage cells of facing racks. The racks are constructed with the base plates extending beyond the edge of the cells which assures that the minimum spacing between storage racks is maintained under all credible conditions. This water-gap flux-trap is larger than those between Region 1 cells, and thus, will act to reduce the reactivity below the cited maximum.

4.6 REGION 2 CRITICALITY ANALYSES AND TOLERANCES

4.6.1 Nominal Design Case

For the nominal storage cell design in Region 2, the criticality safety analyses are summarized in Table 4.2.3. These data confirm that the maximum reactivity in Region 2 remains conservatively less than the regulatory limit (k_{eff} 0.95). An independent calculation with the KENO5a code provides confirmation of the validity of the reference MCNP4a calculations.

4.6.2 Uncertainties Due to Burnup

CASMO-3 was used for the depletion analysis and the restart option was used to analytically transfer the spent fuel into the storage rack configuration at a reference temperature of 4 C (corresponding to the highest reactivity, see Section 4.8.1). Calculations were also made for fuel of several different initial enrichments and interpolated to define the burnup-dependent equivalent enrichments , at each burnup. MCNP4a calculations were then made for the equivalent enrichment to establish the limiting k_{eff} value, which includes all applicable uncertainties and the effect of the axial burnup distribution. These calculations were used to define the boundary of the acceptable domain shown in Figure 4.1.2.

4.6.3 Uncertainties Due to Tolerances

The reactivity effects of manufacturing tolerances are tabulated, along with the tolerances, in Table 4.6.1. The individual reactivity allowances were conservatively calculated for the design basis fresh unburned fuel assembly.

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† The (reactivity) equivalent enrichment is the fresh un-burned fuel enrichment that yields the same reactivity as the depleted fuel, both evaluated in the storage rack configuration. The equivalent enrichment may then be used in three-dimensional MCNP4a or KENO5a calculations.

4.6.4 Eccentric Fuel Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell. However, calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). These calculations indicated that the reactivity effect is small and negative. Therefore, the reference case in which the fuel assemblies are centered is controlling and no uncertainty for eccentricity is necessary.

4.6.5 Water-Gap Spacing Between Racks

The minimum water-gap between racks, which is 0.50 inches between neighboring Region 2 style racks and 1.5 inches between Region 1 and Region 2 style racks, constitutes a neutron flux-trap for the storage cells of facing racks. The racks are constructed with the base plates extending beyond the edge of the cells which assures that the minimum spacing between storage racks is maintained under all credible conditions. Region 2 style racks do not contain water gaps, and thus, this water-gap flux-trap will act to reduce the reactivity below the cited maximum.

4.7 REGION 3 CRITICALITY ANALYSES AND TOLERANCES

4.7.1 Nominal Design Case

For the nominal storage cell design in Region 3, the criticality safety analyses are summarized in Table 4.2.5. These data confirm that the maximum reactivity in Region 3 remains conservatively less than the regulatory limit (k_{eff} 0.95). Independent calculations with the MCNP4a and KENO5a codes provide confirmation of the validity of the reference CASMO-3 calculations.

4.7.2 Uncertainties Due to Burnup

CASMO-3 was used for the depletion and decay time analyses and the restart option was used to analytically transfer the spent fuel into the storage rack configuration at a reference temperature of 160 F (corresponding to the highest reactivity, see Section 4.8.1). Calculations were also made for fuel of several different initial enrichments and interpolated to define the burnup-dependent equivalent enrichments, at each burnup. KENO5a calculations were then made for the equivalent enrichments to determine the effect of the axial burnup distribution. These calculations were made for each of the cooling times. CASMO-3 calculations were used for the establish the limiting k_{eff} value, which includes all applicable uncertainties and the effect of the axial burnup distribution. These calculations were used to define the boundary of the acceptable domains shown in Figures 4.1.3.

4.7.3 Uncertainties Due to Tolerances

The reactivity effects of manufacturing tolerances were calculated for various burnups and each of the defined cooling times with the design basis fuel assembly. For conservatism, the largest reactivity effect for each tolerance was used to establish the corresponding reactivity allowance. These values are tabulated, along with the tolerances, in Table 4.7.1.

4.7.4 Eccentric Fuel Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell. However, calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). Because no credit is taken for the Boraflex panels in the Region 3 racks, these calculations determined that the reactivity effect is small and positive. Therefore, the positive uncertainty associated with fuel eccentricity is included in the determination of the maximum reactivity in Table 4.2.5.

4.8 ABNORMAL AND ACCIDENT CONDITIONS

4.8.1 Temperature and Water Density Effects

4.8.1.1 Region 1 and 2

The moderator temperature coefficient of reactivity in Region 1 and Region 2 is negative. Therefore, a moderator temperature of 4 C (39 F) was assumed for the reference calculations, which assures that the true reactivity will always be lower over the expected range of water temperatures. Temperature effects on reactivity have been calculated (CASMO-3) and the results are shown in Table 4.8.1. In addition, the introduction of voids in the water internal to the storage cell (to simulate boiling) decreased reactivity, as shown in Table 4.8.1.

With soluble boron present, the temperature coefficients of reactivity would differ from those listed in Table 4.8.1. However, the reactivities would also be substantially lower at all temperatures with soluble boron present. The data in Table 4.8.1 is pertinent to the higher-reactivity unborated case.

Since the Monte Carlo codes, MCNP4a and KENO5a, cannot handle temperature dependence, all MCNP4a and KENO5a calculations were performed at 20°C and a positive temperature correction factor (the value of Δk between calculations at 20°C and 4°C) was applied to the results.

4.8.1.2 Region 3

With the assumption that the Boraflex panels are replaced by water, the moderator temperature coefficient of reactivity in Region 3 is positive. Therefore, a moderator temperature of 160 F was assumed for the reference calculations (for normal conditions). Temperatures above 160 F are accident conditions, during which credit for soluble boron is allowed. Temperature effects on reactivity have been calculated (CASMO-3) and the results are shown in Table 4.8.2. In addition, the introduction of voids in the water internal to the storage cell (to simulate boiling) increased reactivity, as shown in Table 4.8.2. Calculations indicate that 100 ppm soluble boron is more than adequate to assure that the limiting k_{eff} of 0.945 is not exceeded for temperatures greater than 160

F and boiling. However, since the spent fuel pool cooling system is capable of maintaining fuel pool water temperature less than 160 F, this condition is outside of the design basis, and no further action is necessary.

With soluble boron present, the temperature coefficients of reactivity would differ from those listed in Table 4.8.2. However, the reactivities would also be substantially lower at all temperatures with soluble boron present. The data in Table 4.8.2 is pertinent to the higher-reactivity unborated case.

The CASMO-3 calculations were performed at 4°C and a positive temperature correction factor (the value of Δk between calculations at 4°C and 160°F) was applied to the results.

4.8.2 Lateral Rack Movement

Lateral motion of the storage racks under seismic conditions could potentially alter the spacing between racks. In Region 1, the minimum water gap between racks (1.5 inches, as limited by the base plate extensions) is larger than the corresponding design water-gap spacing (0.79 inches in one direction and 1.244 inches in the other direction) internal to the racks. Consequently, there will be no positive effect on reactivity.

Region 2 storage cells do not use a flux-trap, and thus, the calculated maximum reactivity does not rely on spacing between racks. Nevertheless, the minimum water gap between Region 2 racks (0.50 inches, as limited by the base plate extensions) and the Boral panels, which are installed on all exterior walls of Region 2 racks, assure that the reactivity is always less than the design limitation. Furthermore, soluble poison would assure that a reactivity less than the design limitation is maintained under all accident or abnormal conditions.

The minimum distance between Region 3 and Region 1 racks is 76.09 inches, and thus, lateral rack moment is of no concern. The minimum water gap between Region 3 and Region 2 racks is 1.28 inches, which is comparable to the water-gap spacing (1.26 inches) internal to the Region 3 racks. In addition, the Region 2 racks have Boral panels installed on all exterior walls (Region 3 racks are assumed to be unpoisoned). Furthermore, soluble poison would assure that a reactivity less than the design limitation is maintained under all accident or abnormal conditions.

4.8.3 Abnormal Location of a Fuel Assembly

The abnormal location of a fresh un-irradiated fuel assembly of 5.0 wt% ^{235}U enrichment could, in the absence of soluble poison, result in exceeding the regulatory limit (k_{eff} 0.95). This could occur if a fresh fuel assembly of the highest permissible enrichment were to be inadvertently loaded into either a Region 2 or Region 3 storage cell. Calculations confirmed that the highest reactivity, including uncertainties, for the worst case postulated accident condition (fresh fuel assembly in Region 3) would exceed the limit on reactivity in the absence of soluble boron. Soluble boron in the spent fuel pool water, for which credit is permitted under these accident conditions, would assure that the reactivity is maintained substantially less than the design limitation. Calculations indicate that the 800 ppm soluble boron, that is to be required by the Technical Specifications during fuel handling operations, is more than adequate to assure that the limiting k_{eff} of 0.945 is not exceeded.

4.8.4 Dropped Fuel Assembly

For the case in which a fuel assembly is assumed to be dropped on top of a rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel in the rack of more than 12 inches, including the potential deformation under seismic or accident conditions. At this separation distance, the effect on reactivity is insignificant. Furthermore, the soluble boron in the pool water assures that the true reactivity is always less than the limiting value for this dropped fuel accident.

4.9 REFERENCES

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Table 4.1.1

Fuel Assembly Specifications

Fuel Rod Data		
Assembly type	Westinghouse Standard	Westinghouse Vantage-5H
Fuel pellet outside diameter, in.	[redacted]	[redacted]
Cladding thickness, in.	[redacted]	[redacted]
Cladding outside diameter, in.	[redacted]	[redacted]
Cladding material	[redacted]	[redacted]
Pellet density, % T.D.	[redacted]	[redacted]
Maximum nominal enrichment, wt% ^{235}U	5.0	5.0
Fuel Assembly Data		
Fuel rod array	17 x 17	17 x 17
Number of fuel rods	[redacted]	[redacted]
Fuel rod pitch, in.	[redacted]	[redacted]
Number of control rod guide and instrument thimbles	[redacted]	[redacted]
Thimble outside diameter, in.	[redacted]	[redacted]
Thimble thickness, in.	[redacted]	[redacted]
Active fuel Length, in.	[redacted]	[redacted]

Table 4.2.1

Summary of the Criticality Safety Analyses for Region 1

Storage Arrangement	3-out-of-4	4-out-of-4
Design Basis Burnups at 5.0 wt% ^{235}U	0	8.0 MWd/kgU
Uncertainties		
Bias Uncertainty (95%/95%)	± 0.0011	± 0.0011
Calculational Statistics‡ (95%/95%, 2.0×σ)	± 0.0011	± 0.0015
Depletion Uncertainty	N/A	± 0.0028
Fuel Eccentricity	negative	negative
Manufacturing Tolerances (Table 4.5.1)	± 0.0111	± 0.0111
Statistical Combination of Uncertainties†	± 0.0112	± 0.0116
Reference k_{eff} (MCNP4a)	0.9122	0.9132
Total Uncertainty (above)	0.0112	0.0116
Axial Burnup Distribution	N/A	negligible
Calculational Bias (see Appendix A)	0.0009	0.0009
Temperature Correction to 4°C (39°F)	0.0015	0.0015
Maximum k_{eff}	**0.9258**	**0.9272††**
Regulatory Limiting k_{eff}	**0.9500**	**0.9500**

‡ The value used for the MCNP4a (or KENO5a) statistical uncertainty is 2.0 times the estimated standard deviation. Each final k value calculated by MCNP4a (or KENO5a) is the result of averaging a minimum of 200 cycle k values, and thus, is based on a minimum sample size of 200. The K multiplier, for a one-sided statistical tolerance with 95% probability at the 95% confidence level, corresponding to a sample size of 200, is 1.84. However, for this analysis a value of 2.0 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 200.

† Square root of the sum of the squares.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9270.

Table 4.2.2

Burnup-Enrichment Limits in Region 1

Nominal Initial Fuel Enrichment (wt% ^{235}U)	Minimum Fuel Burnup (MWd/kgU)
3.7	0.00
4.0	1.91
4.5	5.12
5.0	8.00

Table 4.2.3

Summary of the Criticality Safety Analyses for Region 2

Design Basis Burnup at 5.0 wt% ^{235}U	39.0 MWd/kgU
Uncertainties	
Bias Uncertainty (95%/95%)	± 0.0011
Calculational Statistics‡ (95%/95%, 2.0×σ)	± 0.0013
Depletion Uncertainty	± 0.0142
Fuel Eccentricity	negative
Manufacturing Tolerances (Table 4.6.1)	± 0.0059
Statistical Combination of Uncertainties†	± 0.0155
Reference k_{eff} (MCNP4a)	0.9142
Total Uncertainty (above)	0.0155
Axial Burnup Distribution	0.0110
Calculational Bias (see Appendix A)	0.0009
Temperature Correction to 4°C (39°F)	0.0020
Maximum k_{eff}	**0.9436††**
Regulatory Limiting k_{eff}	**0.9500**

‡ The value used for the MCNP4a (or KENO5a) statistical uncertainty is 2.0 times the estimated standard deviation. Each final k value calculated by MCNP4a (or KENO5a) is the result of averaging a minimum of 200 cycle k values, and thus, is based on a minimum sample size of 200. The K multiplier, for a one-sided statistical tolerance with 95% probability at the 95% confidence level, corresponding to a sample size of 200, is 1.84. However, for this analysis a value of 2.0 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 200.

† Square root of the sum of the squares.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9449.

Table 4.2.4

Burnup-Enrichment Limits in Region 2

Nominal Initial Fuel Enrichment (wt% ^{235}U)	Minimum Fuel Burnup (MWd/kgU)
2.0	3.48
2.5	10.04
3.0	15.92
3.5	21.48
4.0	26.83
4.5	33.75
5.0	39.00

Table 4.2.5

Summary of the Criticality Safety Analyses for Region 3

Cooling Time (years)	0	5	10	20
Design Basis Burnup (MWd/kgU) at 5.0 wt% ^{235}U	55.41	49.90	47.31	43.91
Uncertainties				
Depletion Uncertainty	0.0182	0.0186	0.0189	0.0189
Fuel Eccentricity	0.0017	0.0017	0.0017	0.0017
Manufacturing Tolerances (Table 4.7.1)	[redacted]	[redacted]	[redacted]	[redacted]
Statistical Combination of Uncertainties†	[redacted]	[redacted]	[redacted]	[redacted]
Reference k_∞ (CASMO-3)	0.8796	0.8705	0.8643	0.8650
Total Uncertainty (above)	0.0196	0.0200	0.0203	0.0203
Axial Burnup Distribution	0.0298	0.0386	0.0445	0.0438
Temperature Correction to 160°F	0.0160	0.0160	0.0160	0.0160
Maximum k_{eff}	**0.945**	**0.945**	**0.945**	**0.945**
Regulatory Limiting k_{eff}	**0.950**	**0.950**	**0.950**	**0.950**

† Square root of the sum of the squares.

Table 4.2.6

Burnup-Enrichment Limits in Region 3 for Various Decay Times

	Minimum Fuel Burnup (MWd/kgU)			
Nominal Initial Fuel Enrichment (wt% ^{235}U)	0 (years decay time)	5 (years decay time)	10 (years decay time)	20 (years decay time)
2.0	10.64	9.64	9.05	8.47
2.5	18.51	16.55	15.42	14.32
3.0	25.62	22.66	21.08	19.56
3.5	32.58	28.44	26.50	24.59
4.0	40.33	35.39	32.82	30.63
4.5	47.95	42.67	40.03	37.25
5.0	55.41	49.90	47.31	43.91

Table 4.2.7

Reactivity Effects of Abnormal and Accident Conditions in Regions 1 and 2

Abnormal/Accident Conditions	Reactivity Effect
Temperature Increase (above 4°C)	Negative (Table 4.8.1)
Void (boiling)	Negative (Table 4.8.1)
Assembly Drop (on top of rack)	Negligible
Lateral Rack Movement	Negligible
Misplacement of a fresh fuel assembly	Positive - controlled by less than 800 ppm soluble boron (a minimum 800 ppm soluble boron is to be required by Technical Specifications during fuel movement)

Table 4A.5

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH THICK LEAD AND STEEL REFLECTORS[†]

Ref.	Case	E, wt%	Separation, cm	MCNP4a k_{eff}	KENO5a k_{eff}
4A.11	Steel Reflector	2.35	1.321	0.9980±0.0009	0.9992±0.0006
		2.35	2.616	0.9968±0.0009	0.9964±0.0006
		2.35	3.912	0.9974±0.0010	0.9980±0.0006
		2.35	∞	0.9962±0.0008	0.9939±0.0006
4A.11	Steel Reflector	4.306	1.321	0.9997±0.0010	1.0012±0.0007
		4.306	2.616	0.9994±0.0012	0.9974±0.0007
		4.306	3.405	0.9969±0.0011	0.9951±0.0007
		4.306	∞	0.9910±0.0020	0.9947±0.0007
4A.12	Lead Reflector	4.306	0.55	1.0025±0.0011	0.9997±0.0007
		4.306	1.956	1.0000±0.0012	0.9985±0.0007
		4.306	5.405	0.9971±0.0012	0.9946±0.0007

† Arranged in order of increasing reflector-fuel spacing.

Table 4.2.8

Reactivity Effects of Abnormal and Accident Conditions in Region 3

Abnormal/Accident Conditions	Reactivity Effect
Temperature Increase (above 160°F)	Positive (Table 4.8.2) - controlled by 100 ppm soluble boron (however, outside of design basis)
Void (boiling)	Positive (Table 4.8.2) - controlled by 100 ppm soluble boron (however, outside of design basis)
Assembly Drop (on top of rack)	Negligible
Lateral Rack Movement	Negligible
Misplacement of a fresh fuel assembly	Positive - controlled by less than 800 ppm soluble boron (a minimum 800 ppm soluble boron is to be required by Technical Specifications during fuel movement)

Table 4.5.1

Reactivity Effects of Manufacturing Tolerances in Region 1

Tolerance	Reactivity Effect, Δk
Minimum Boral loading ([redacted]nominal)	±0.0029
Minimum Boral width [redacted]nominal)	±0.0018
Maximum box I.D. [redacted]nominal)	±0.0102
Maximum box wall thickness [redacted]' nominal)	±0.0007
Density tolerance [redacted]nominal)	±0.0018
Enrichment [redacted] nominal)	±0.0017
Total (statistical sum)†	**±0.0111**

† Square root of the sum of the squares.

Table 4.6.1

Reactivity Effects of Manufacturing Tolerances in Region 2

Tolerance	Reactivity Effect, Δk
Minimum Boral loading ([redacted]nominal)	±0.0045
Minimum Boral width ([redacted]nominal)	±0.0023
Minimum box I.D. ([redacted]nominal)	±0.0017
Maximum box wall thickness [redacted]' nominal)	±0.0002
Density ([redacted]nominal)	±0.0013
Enrichment [redacted]ominal)	±0.0021
Total (statistical sum)†	**±0.0059**

† Square root of the sum of the squares.

Table 4.7.1

Reactivity Effects of Manufacturing Tolerances in Region 3

Tolerance	Reactivity Effect, Δk
Minimum box I.D. ([redacted] nominal)	±0.0004
Minimum pitch ([redacted] nominal)	±0.0030
Minimum box wall thickness ([redacted] nominal)	±0.0026
Minimum sheathing thickness ([redacted] nominal)	±0.0033
Density ([redacted] nominal)	±0.0032
Enrichment [redacted] nominal)	±0.0039
Total (statistical sum)†	**±0.0072**

† Square root of the sum of the squares.

Table 4.8.1

Reactivity Effects of Temperature and Void in Regions 1 and 2

	Reactivity Effect, Δk	
Case	Region 1	Region 2
4°C (39°F)	reference	reference
20°C (68°F)	-0.0015	-0.0020
60°C (140°F)	-0.0084	-0.0094
120°C (248°F)	-0.0241	-0.0253
120°C w/ 10% void	-0.0527	-0.0508

Table 4.8.2

Reactivity Effects of Temperature and Void in Region 3

Case	Reactivity Effect, Δk
4°C (39°F)	-0.0160
20°C (68°F)	-0.0122
40°C (104°F)	0.0074
65°C (149°F)	-0.0014
71.1°C (160°F)	reference
90°C (194°F)	+0.0045
120°C (248°F)	+0.0123
120°C w/ 10% void	+0.0163

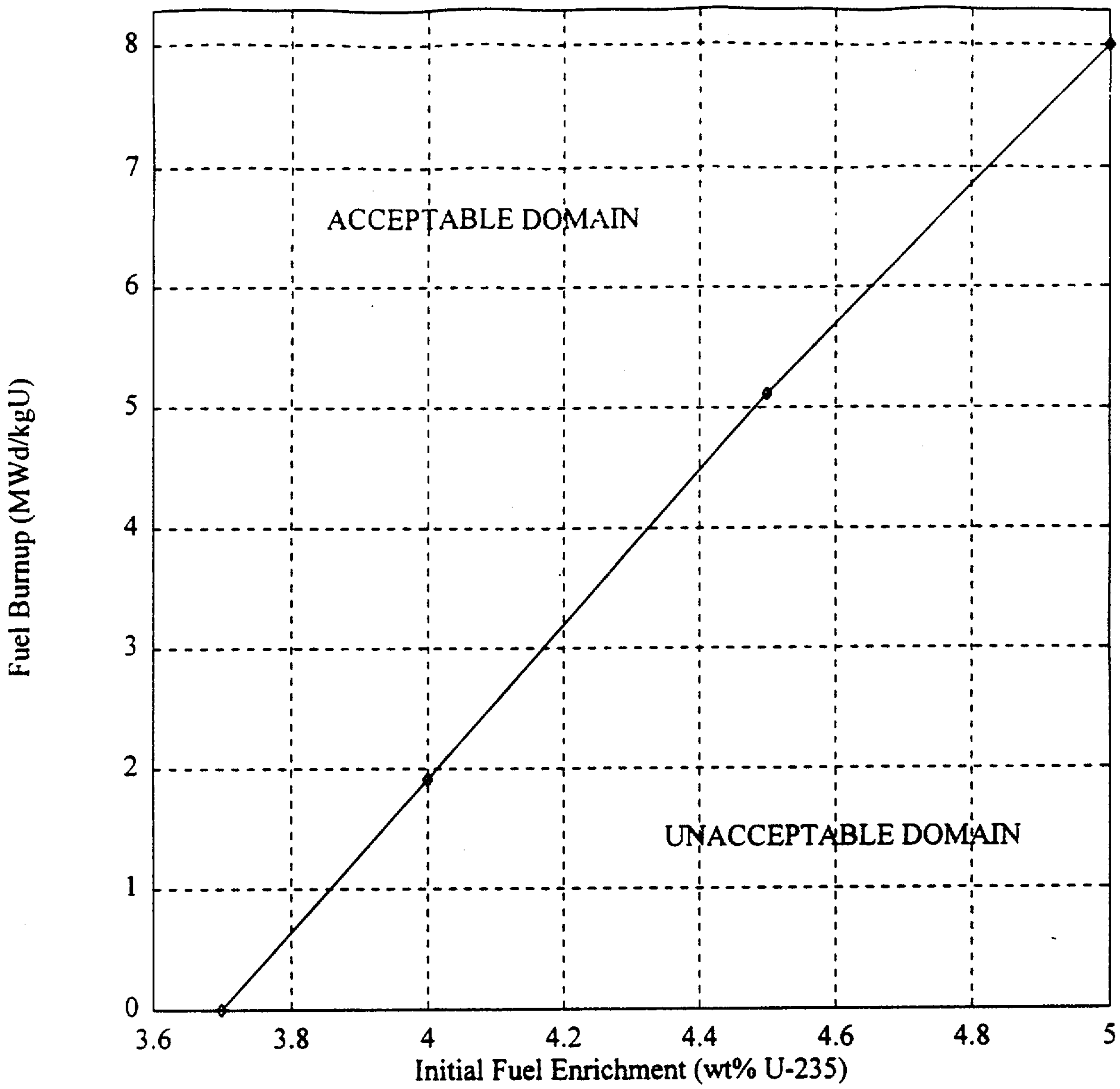

Figure 4.1.1 Minimum Required Fuel Assembly Burnup as a Function of Nominal Initial Enrichment to Permit Unrestricted Storage in Region 1

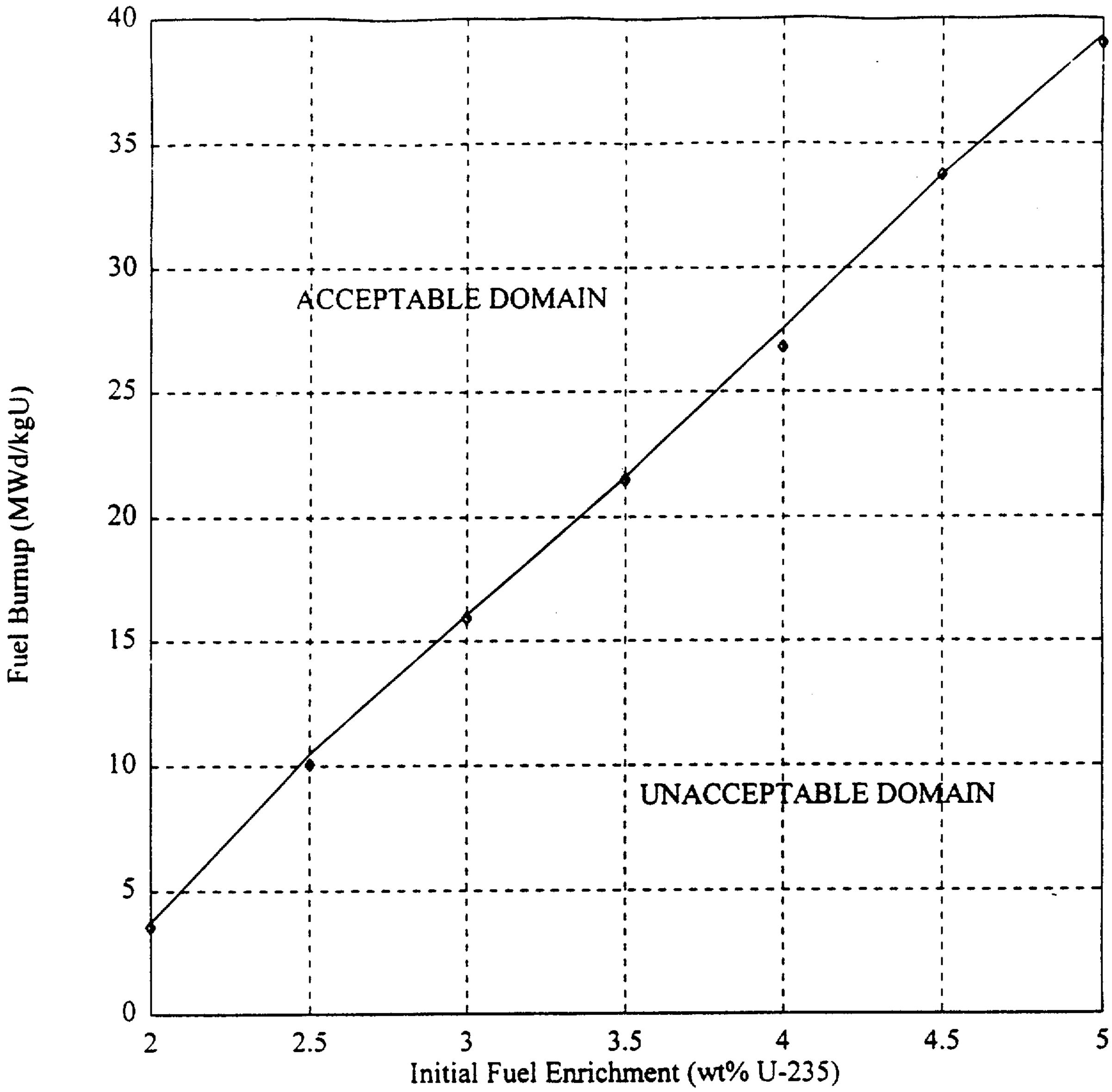

Figure 4.1.2 Minimum Required Fuel Assembly Burnup as a Function of Nominal Initial Enrichment to Permit Unrestricted Storage in Region 2 (Fuel assemblies with enrichments less than 2.0 wt% ^{235}U will conservatively be required to meet the burnup requirements of 2.0 wt% ^{235}U assemblies).

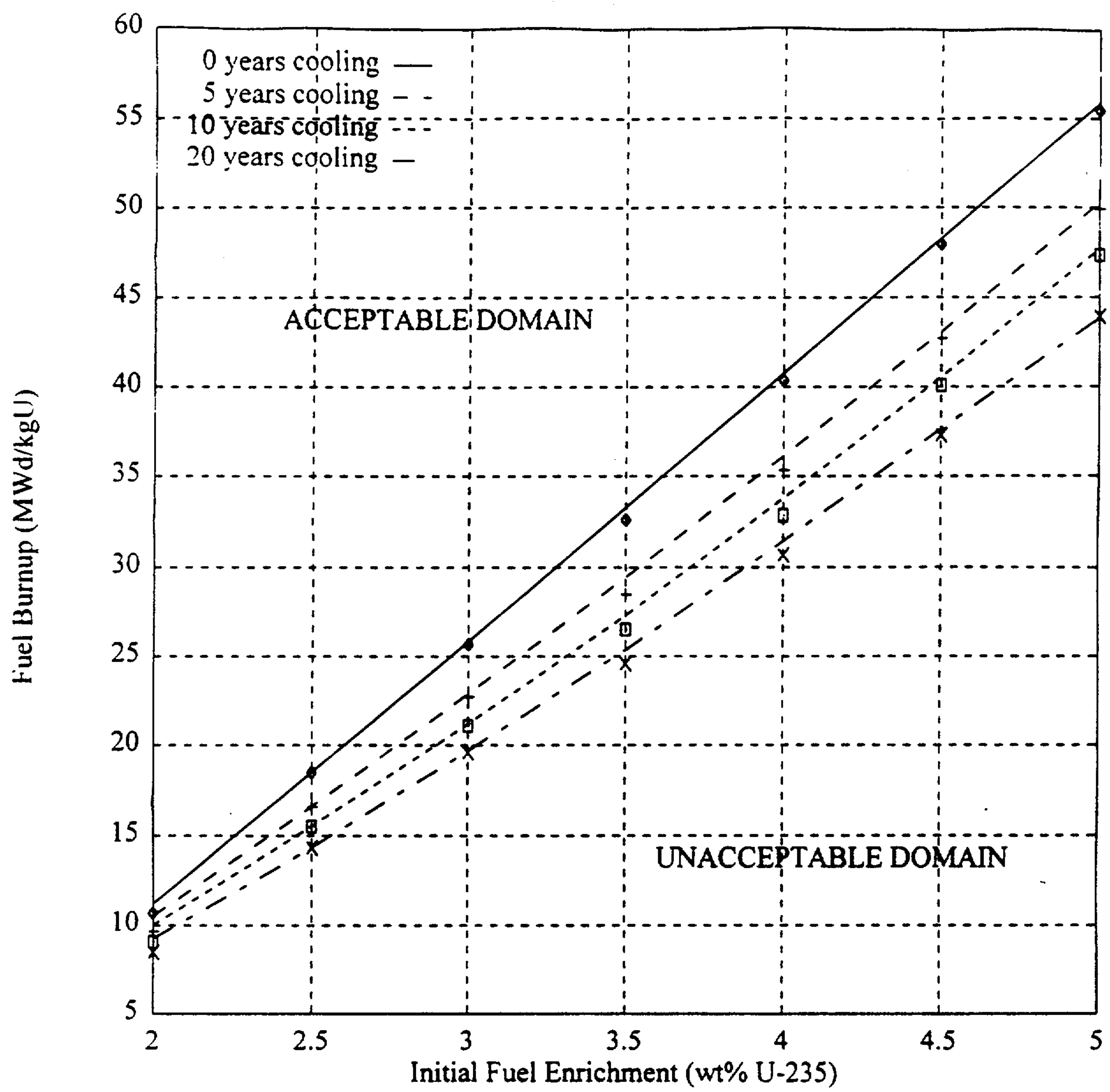

Figure 4.1.3 Minimum Required Fuel Assembly Burnup as a Function of Nominal Initial Enrichment to Permit Unrestricted Storage in Region 3 (Fuel assemblies with enrichments less than 2.0 wt% ^{235}U will conservatively be required to meet the burnup requirements of 2.0 wt% ^{235}U assemblies).

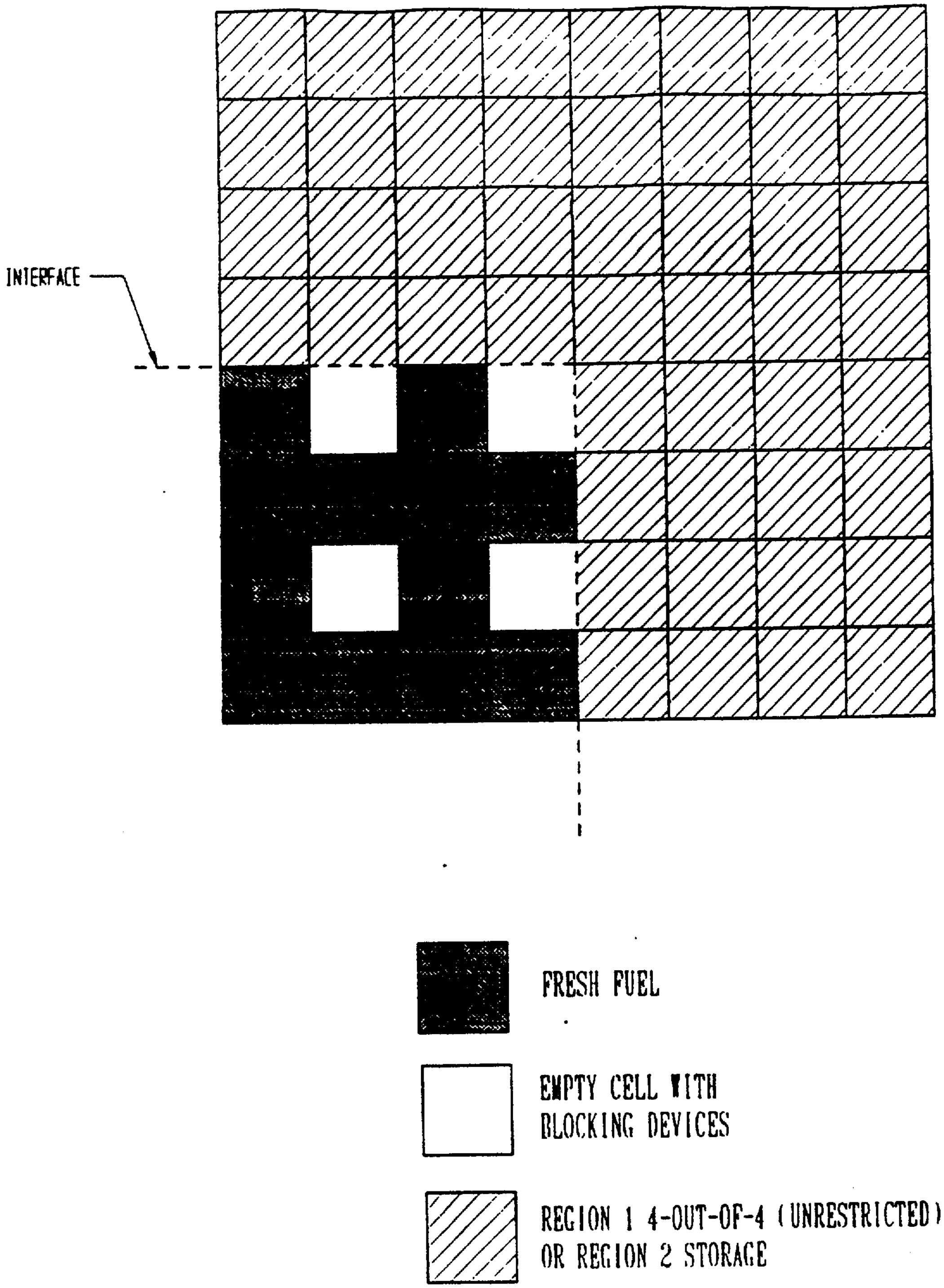

Figure 4.2.1 Illustration of the Interface Requirement Between 3-out-of-4 and 4-out-of-4 (Unrestricted) Storage in Region 1

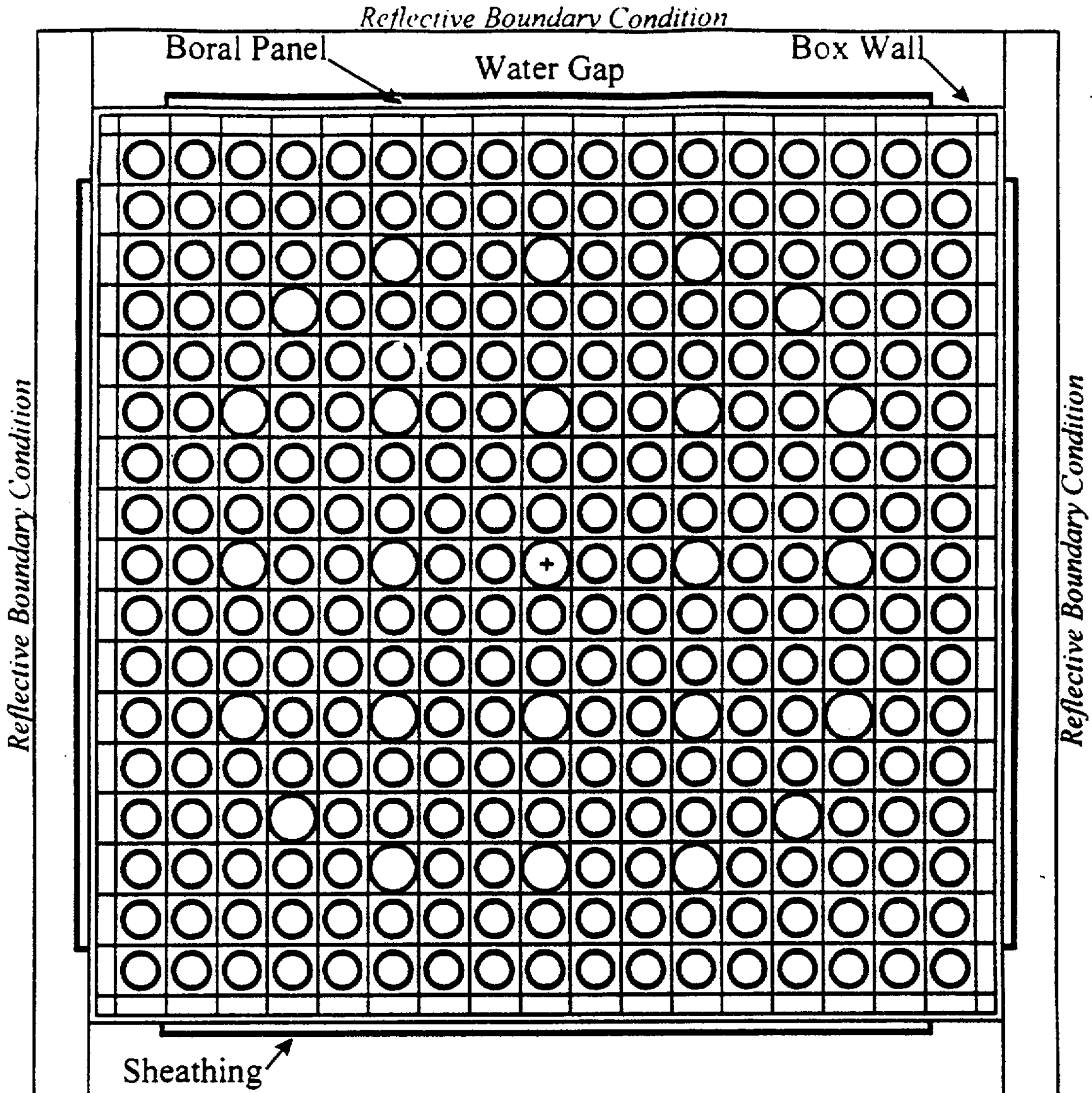

Figure 4.3.1 A Two-Dimensional Representation of the Calculational Model Used for the Region 1 Rack Analysis. This Figure was Drawn with the Two-Dimensional Plotter in MCNP4a

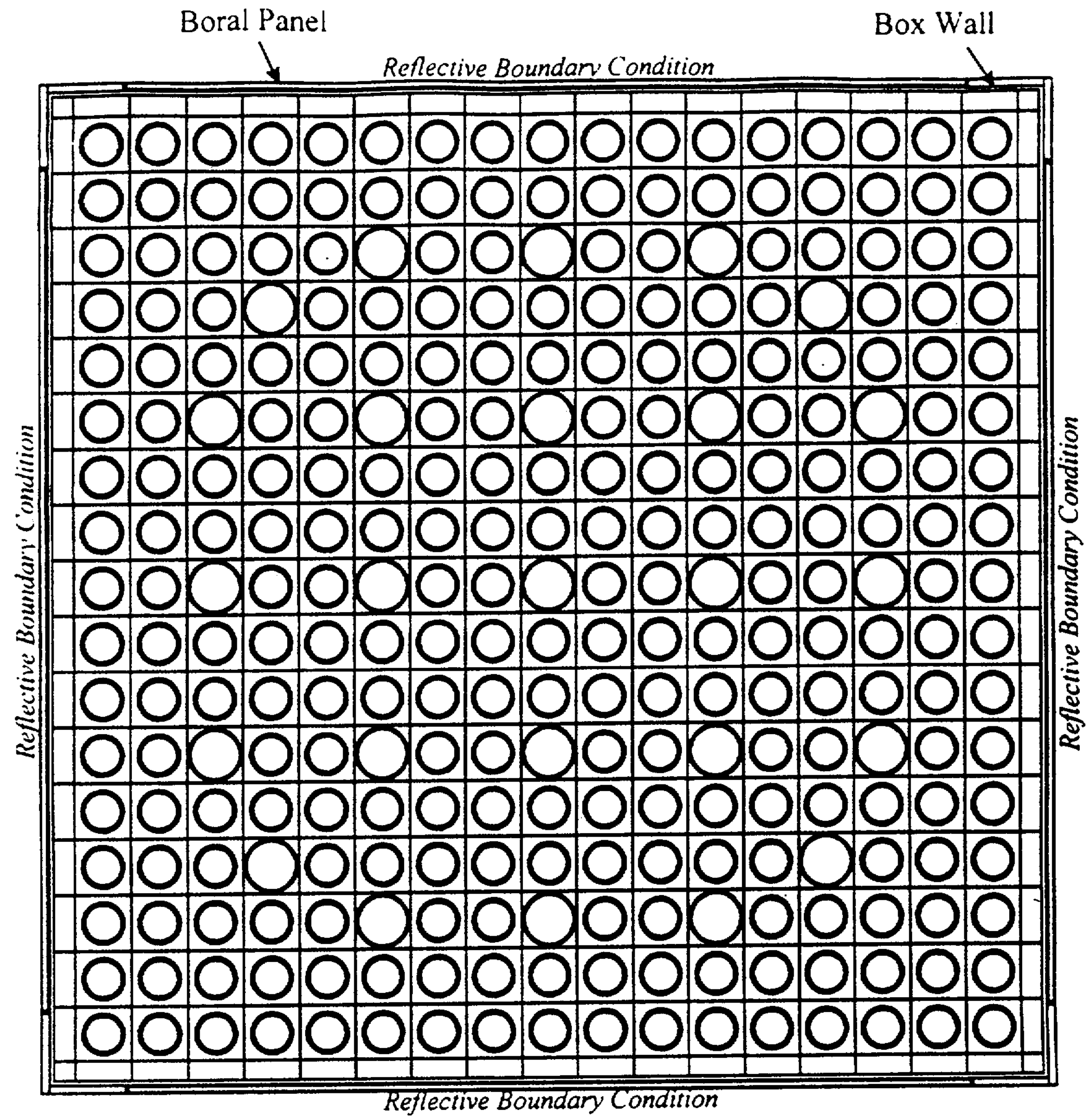

Figure 4.3.2 A Two-Dimensional Representation of the Calculational Model Used for the Region 2 Rack Analysis. This Figure was Drawn with the Two-Dimensional Plotter in MCNP4a

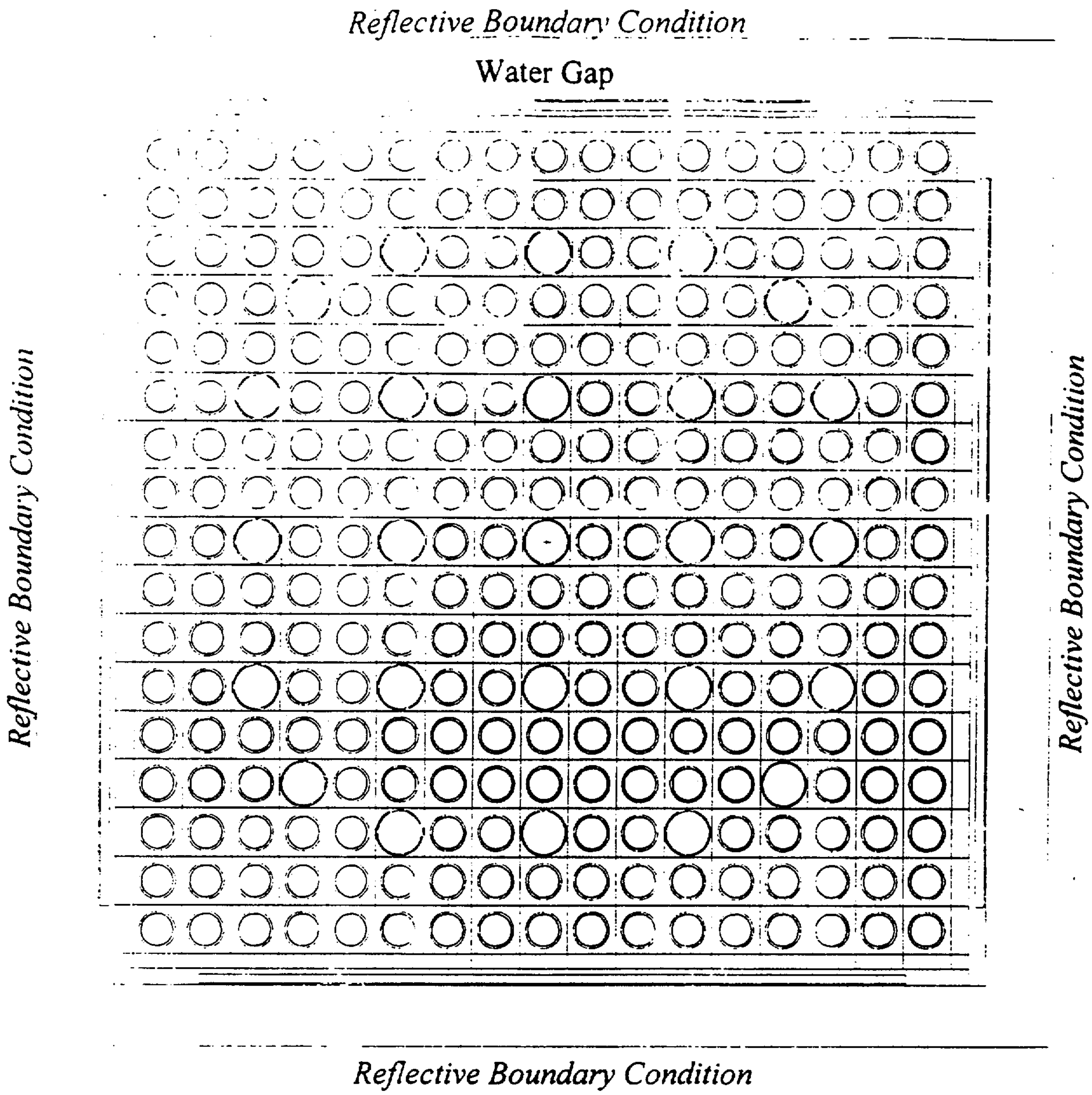

Figure 4.3.3 A Two-Dimensional Representation of the Calculational Model Used for the Region 3 Rack Analysis. This Figure was Drawn with the Two-Dimensional Plotter in MCNP4a

APPENDIX 4A: BENCHMARK CALCULATIONS

4A.1 INTRODUCTION AND SUMMARY

Benchmark calculations have been made on selected critical experiments, chosen, in so far as possible, to bound the range of variables in the rack designs. Two independent methods of analysis were used, differing in cross section libraries and in the treatment of the cross sections. MCNP4a [4A.1] is a continuous energy Monte Carlo code and KENO5a [4A.2] uses group-dependent cross sections. For the KENO5a analyses reported here, the 238-group library was chosen, processed through the NITAWL-II [4A.2] program to create a working library and to account for resonance self-shielding in uranium-238 (Nordheim integral treatment). The 238 group library was chosen to avoid or minimize the errors[†] (trends) that have been reported (e.g., [4A.3 through 4A.5]) for calculations with collapsed cross section sets.

In rack designs, the three most significant parameters affecting criticality are (1) the fuel enrichment, (2) the ^{10}B loading in the neutron absorber, and (3) the lattice spacing (or water-gap thickness if a flux-trap design is used). Other parameters, within the normal range of rack and fuel designs, have a smaller effect, but are also included in the analyses.

Table 4A.1 summarizes results of the benchmark calculations for all cases selected and analyzed, as referenced in the table. The effect of the major variables are discussed in subsequent sections below. It is important to note that there is obviously considerable overlap in parameters since it is not possible to vary a single parameter and maintain criticality; some other parameter or parameters must be concurrently varied to maintain criticality.

One possible way of representing the data is through a spectrum index that incorporates all of the variations in parameters. KENO5a computes and prints the "energy of the average lethargy causing fission" (EALF). In MCNP4a, by utilizing the tally option with the identical 238-group energy structure as in KENO5a, the number of fissions in each group may be collected and the EALF determined (post-processing).

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† Small but observable trends (errors) have been reported for calculations with the 27-group and 44-group collapsed libraries. These errors are probably due to the use of a single collapsing spectrum when the spectrum should be different for the various cases analyzed, as evidenced by the spectrum indices.

Figures 4A.1 and 4A.2 show the calculated k_{eff} for the benchmark critical experiments as a function of the EALF for MCNP4a and KENO5a, respectively (UO_2 fuel only). The scatter in the data (even for comparatively minor variation in critical parameters) represents experimental error† in performing the critical experiments within each laboratory, as well as between the various testing laboratories. The B&W critical experiments show a larger experimental error than the PNL criticals. This would be expected since the B&W criticals encompass a greater range of critical parameters than the PNL criticals.

Linear regression analysis of the data in Figures 4A.1 and 4A.2 show that there are no trends, as evidenced by very low values of the correlation coefficient (0.13 for MCNP4a and 0.21 for KENO5a). The total bias (systematic error, or mean of the deviation from a k_{eff} of exactly 1.000) for the two methods of analysis are shown in the table below.

Calculational Bias of MCNP4a and KENO5a	
MCNP4a	0.0009 ± 0.0011
KENO5a	0.0030 ± 0.0012

The bias and standard error of the bias were derived directly from the calculated k_{eff} values in Table 4A.1 using the following equations††, with the standard error multiplied by the one-sided K-factor for 95% probability at the 95% confidence level from NBS Handbook 91 [4A.18] (for the number of cases analyzed, the K-factor is ~2.05 or slightly more than 2).

$$\bar{k} = \frac{1}{n} \sum_{i}^{n} k_i \qquad (4A.1)$$

† A classical example of experimental error is the corrected enrichment in the PNL experiments, first as an addendum to the initial report and, secondly, by revised values in subsequent reports for the same fuel rods.

†† These equations may be found in any standard text on statistics, for example, reference [4A.6] (or the MCNP4a manual) and is the same methodology used in MCNP4a and in KENO5a.

$$\sigma_{\bar{k}}^2 = \frac{\sum_{i=1}^{n} k_i^2 - (\sum_{i=1}^{n} k_i)^2 / n}{n\,(n-1)} \tag{4A.2}$$

$$Bias = (1 - \bar{k}) \pm K\, \sigma_{\bar{k}} \tag{4A.3}$$

where k_i are the calculated reactivities of n critical experiments; $\sigma_{\bar{k}}$ is the unbiased estimator of the standard deviation of the mean (also called the standard error of the bias (mean)); K is the one-sided multiplier for 95% probability at the 95% confidence level (NBS Handbook 91 [4A.18]).

Formula 4.A.3 is based on the methodology of the National Bureau of Standards (now NIST) and is used to calculate the values presented on page 4.A-2. The first portion of the equation, ($1-\bar{k}$), is the actual bias which is added to the MCNP4a and KENO5a results. The second term, $K\sigma_{\bar{k}}$, is the uncertainty or standard error associated with the bias. The K values used were obtained from the National Bureau of Standards Handbook 91 and are for one-sided statistical tolerance limits for 95% probability at the 95% confidence level. The actual K values for the 56 critical experiments evaluated with MCNP4a and the 53 critical experiments evaluated with KENO5a are 2.04 and 2.05, respectively.

The bias values are used to evaluate the maximum k_{eff} values for the rack designs. KENO5a has a slightly larger systematic error than MCNP4a, but both result in greater precision than published data [4A.3 through 4A.5] would indicate for collapsed cross section sets in KENO5a (SCALE) calculations.

4A.2 Effect of Enrichment

The benchmark critical experiments include those with enrichments ranging from 2.46 w/o to 5.74 w/o and therefore span the enrichment range for rack designs. Figures 4A.3 and 4A.4 show the calculated k_{eff} values (Table 4A.1) as a function of the fuel enrichment reported for the critical experiments. Linear regression analyses for these data confirms that there are no trends, as indicated by low values of the correlation coefficients (0.03 for MCNP4a and 0.38 for KENO5a). Thus, there are no corrections to the bias for the various enrichments.

As further confirmation of the absence of any trends with enrichment, a typical configuration was calculated with both MCNP4a and KENO5a for various enrichments. The cross-comparison of calculations with codes of comparable sophistication is suggested in Reg. Guide 3.41. Results of this comparison, shown in Table 4A.2 and Figure 4A.5, confirm no significant difference in the calculated values of k_{eff} for the two independent codes as evidenced by the 45° slope of the curve. Since it is very unlikely that two independent methods of analysis would be subject to the same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias.

4A.3 Effect of ^{10}B Loading

Several laboratories have performed critical experiments with a variety of thin absorber panels similar to the Boral panels in the rack designs. Of these critical experiments, those performed by B&W are the most representative of the rack designs. PNL has also made some measurements with absorber plates, but, with one exception (a flux-trap experiment), the reactivity worth of the absorbers in the PNL tests is very low and any significant errors that might exist in the treatment of strong thin absorbers could not be revealed.

Table 4A.3 lists the subset of experiments using thin neutron absorbers (from Table 4A.1) and shows the reactivity worth (Δk) of the absorber.†

No trends with reactivity worth of the absorber are evident, although based on the calculations shown in Table 4A.3, some of the B&W critical experiments seem to have unusually large experimental errors. B&W made an effort to report some of their experimental errors. Other laboratories did not evaluate their experimental errors.

To further confirm the absence of a significant trend with ^{10}B concentration in the absorber, a cross-comparison was made with MCNP4a and KENO5a (as suggested in Reg. Guide 3.41). Results are shown in Figure 4A.6 and Table 4A.4 for a typical geometry. These data substantiate the absence of any error (trend) in either of the two codes for the conditions analyzed (data points fall on a 45° line, within an expected 95% probability limit).

† The reactivity worth of the absorber panels was determined by repeating the calculation with the absorber analytically removed and calculating the incremental (Δk) change in reactivity due to the absorber.

4A.4 Miscellaneous and Minor Parameters

4A.4.1 Reflector Material and Spacings

PNL has performed a number of critical experiments with thick steel and lead reflectors.[†] Analysis of these critical experiments are listed in Table 4A.5 (subset of data in Table 4A.1). There appears to be a small tendency toward overprediction of k_{eff} at the lower spacing, although there are an insufficient number of data points in each series to allow a quantitative determination of any trends. The tendency toward overprediction at close spacing means that the rack calculations may be slightly more conservative than otherwise.

4A.4.2 Fuel Pellet Diameter and Lattice Pitch

The critical experiments selected for analysis cover a range of fuel pellet diameters from 0.311 to 0.444 inches, and lattice spacings from 0.476 to 1.00 inches. In the rack designs, the fuel pellet diameters range from 0.303 to 0.3805 inches O.D. (0.496 to 0.580 inch lattice spacing) for PWR fuel and from 0.3224 to 0.494 inches O.D. (0.488 to 0.740 inch lattice spacing) for BWR fuel. Thus, the critical experiments analyzed provide a reasonable representation of power reactor fuel. Based on the data in Table 4A.1, there does not appear to be any observable trend with either fuel pellet diameter or lattice pitch, at least over the range of the critical experiments applicable to rack designs.

4A.4.3 Soluble Boron Concentration Effects

Various soluble boron concentrations were used in the B&W series of critical experiments and in one PNL experiment, with boron concentrations ranging up to 2550 ppm. Results of MCNP4a (and one KENO5a) calculations are shown in Table 4A.6. Analyses of the very high boron concentration experiments (> 1300 ppm) show a tendency to slightly overpredict reactivity for the three experiments exceeding 1300 ppm. In turn, this would suggest that the evaluation of the racks with higher soluble boron concentrations could be slightly conservative.

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† Parallel experiments with a depleted uranium reflector were also performed but not included in the present analysis since they are not pertinent to the Holtec rack design.

4A.5 MOX Fuel

The number of critical experiments with PuO_2 bearing fuel (MOX) is more limited than for UO_2 fuel. However, a number of MOX critical experiments have been analyzed and the results are shown in Table 4A.7. Results of these analyses are generally above a k_{eff} of 1.00, indicating that when Pu is present, both MCNP4a and KENO5a overpredict the reactivity. This may indicate that calculation for MOX fuel will be expected to be conservative, especially with MCNP4a. It may be noted that for the larger lattice spacings, the KENO5a calculated reactivities are below 1.00, suggesting that a small trend may exist with KENO5a. It is also possible that the overprediction in k_{eff} for both codes may be due to a small inadequacy in the determination of the Pu-241 decay and Am-241 growth. This possibility is supported by the consistency in calculated k_{eff} over a wide range of the spectral index (energy of the average lethargy causing fission).

4A.6 References

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[4A.3] M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-G Group ENDF/B-Y Cross-Section Library for Use in Criticality Safety Analyses", NUREG/CR-6102 (ORNL/TM-12460) Oak Ridge National Laboratory, September 1994.

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[4A.5] O.W. Hermann et al., "Validation of the Scale System for PWR Spent Fuel Isotopic Composition Analysis", ORNL-TM-12667, Oak Ridge National Laboratory, undated.

[4A.6] R.J. Larsen and M.L. Marx, An Introduction to Mathematical Statistics and its Applications, Prentice-Hall, 1986.

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[4A.12] S.R. Bierman et al., Criticality Experiments with Subcritical Clusters of 2.35 w/o and 4.31 w/o ^{235}U Enriched UO_2 Rods in Water with Uranium or Lead Reflecting Walls, PNL-3926, Battelle Pacific Northwest Laboratory, December, 1981.

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[4A.14] S.R. Bierman, Criticality Experiments with Neutron Flux Traps Containing Voids, PNL-7167, Battelle Pacific Northwest Laboratory, April 1990.

[4A.15] B.M. Durst et al., Critical Experiments with 4.31 wt % ^{235}U Enriched UO_2 Rods in Highly Borated Water Lattices, PNL-4267, Battelle Pacific Northwest Laboratory, August 1982.

[4A.16] S.R. Bierman, Criticality Experiments with Fast Test Reactor Fuel Pins in Organic Moderator, PNL-5803, Battelle Pacific Northwest Laboratory, December 1981.

[4A.17] E.G. Taylor et al., Saxton Plutonium Program Critical Experiments for the Saxton Partial Plutonium Core, WCAP-3385-54, Westinghouse Electric Corp., Atomic Power Division, December 1965.

[4A.18] M.G. Natrella, <u>Experimental Statistics</u>, National Bureau of Standards, Handbook 91, August 1963.

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF[†] (eV) MCNP4a	EALF[†] (eV) KENO5a
1	B&W-1484 (4A.7)	Core I	2.46	0.9964 ± 0.0010	0.9898± 0.0006	0.1759	0.1753
2	B&W-1484 (4A.7)	Core II	2.46	1.0008 ± 0.0011	1.0015 ± 0.0005	0.2553	0.2446
3	B&W-1484 (4A.7)	Core III	2.46	1.0010 ± 0.0012	1.0005 ± 0.0005	0.1999	0.1939
4	B&W-1484 (4A.7)	Core IX	2.46	0.9956 ± 0.0012	0.9901 ± 0.0006	0.1422	0.1426
5	B&W-1484 (4A.7)	Core X	2.46	0.9980 ± 0.0014	0.9922 ± 0.0006	0.1513	0.1499
6	B&W-1484 (4A.7)	Core XI	2.46	0.9978 ± 0.0012	1.0005 ± 0.0005	0.2031	0.1947
7	B&W-1484 (4A.7)	Core XII	2.46	0.9988 ± 0.0011	0.9978 ± 0.0006	0.1718	0.1662
8	B&W-1484 (4A.7)	Core XIII	2.46	1.0020 ± 0.0010	0.9952 ± 0.0006	0.1988	0.1965
9	B&W-1484 (4A.7)	Core XIV	2.46	0.9953 ± 0.0011	0.9928 ± 0.0006	0.2022	0.1986
10	B&W-1484 (4A.7)	Core XV[††]	2.46	0.9910 ± 0.0011	0.9909 ± 0.0006	0.2092	0.2014
11	B&W-1484 (4A.7)	Core XVI[††]	2.46	0.9935 ± 0.0010	0.9889 ± 0.0006	0.1757	0.1713
12	B&W-1484 (4A.7)	Core XVII	2.46	0.9962 ± 0.0012	0.9942 ± 0.0005	0.2083	0.2021
13	B&W-1484 (4A.7)	Core XVIII	2.46	1.0036 ± 0.0012	0.9931 ± 0.0006	0.1705	0.1708

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF[1] (eV) MCNP4a	EALF[1] (eV) KENO5a
14	B&W-1484 (4A.7)	Core XIX	2.46	0.9961 ± 0.0012	0.9971 ± 0.0005	0.2103	0.2011
15	B&W-1484 (4A.7)	Core XX	2.46	1.0008 ± 0.0011	0.9932 ± 0.0006	0.1724	0.1701
16	B&W-1484 (4A.7)	Core XXI	2.46	0.9994 ± 0.0010	0.9918 ± 0.0006	0.1544	0.1536
17	B&W-1645 (4A.8)	S-type Fuel, w/886 ppm B	2.46	0.9970 ± 0.0010	0.9924 ± 0.0006	1.4475	1.4680
18	B&W-1645 (4A.8)	S-type Fuel, w/746 ppm B	2.46	0.9990 ± 0.0010	0.9913 ± 0.0006	1.5463	1.5660
19	B&W-1645 (4A.8)	SO-type Fuel, w/1156 ppm B	2.46	0.9972 ± 0.0009	0.9949 ± 0.0005	0.4241	0.4331
20	B&W-1810 (4A.9)	Case 1 1337 ppm B	2.46	1.0023 ± 0.0010	NC	0.1531	NC
21	B&W-1810 (4A.9)	Case 12 1899 ppm B	2.46/4.02	1.0060 ± 0.0009	NC	0.4493	NC
22	French (4A.10)	Water Moderator 0 gap	4.75	0.9966 ± 0.0013	NC	0.2172	NC
23	French (4A.10)	Water Moderator 2.5 cm gap	4.75	0.9952 ± 0.0012	NC	0.1778	NC
24	French (4A.10)	Water Moderator 5 cm gap	4.75	0.9943 ± 0.0010	NC	0.1677	NC
25	French (4A.10)	Water Moderator 10 cm gap	4.75	0.9979 ± 0.0010	NC	0.1736	NC
26	PNL-3602 (4A.11)	Steel Reflector, 0 separation	2.35	NC	1.0004 ± 0.0006	NC	0.1018

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF [†] (eV) MCNP4a	EALF [†] (eV) KENO5a
27	PNL-3602 (4A.11)	Steel Reflector, 1.321 cm sepn.	2.35	0.9980 ± 0.0009	0.9992 ± 0.0006	0.1000	0.0909
28	PNL-3602 (4A.11)	Steel Reflector, 2.616 cm sepn	2.35	0.9968 ± 0.0009	0.9964 ± 0.0006	0.0981	0.0975
29	PNL-3602 (4A.11)	Steel Reflector, 3.912 cm sepn.	2.35	0.9974 ± 0.0010	0.9980 ± 0.0006	0.0976	0.0970
30	PNL-3602 (4A.11)	Steel Reflector, infinite sepn.	2.35	0.9962 ± 0.0008	0.9939 ± 0.0006	0.0973	0.0968
31	PNL-3602 (4A.11)	Steel Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3282
32	PNL-3602 (4A.11)	Steel Reflector, 1.321 cm sepn.	4.306	0.9997 ± 0.0010	1.0012 ± 0.0007	0.3016	0.3039
33	PNL-3602 (4A.11)	Steel Reflector, 2.616 cm sepn.	4.306	0.9994 ± 0.0012	0.9974 ± 0.0007	0.2911	0.2927
34	PNL-3602 (4A.11)	Steel Reflector, 5.405 cm sepn.	4.306	0.9969 ± 0.0011	0.9951 ± 0.0007	0.2828	0.2860
35	PNL-3602 (4A.11)	Steel Reflector, Infinite sepn. [††]	4.306	0.9910 ± 0.0020	0.9947 ± 0.0007	0.2851	0.2864
36	PNL-3602 (4A.11)	Steel Reflector, with Boral Sheets	4.306	0.9941 ± 0.0011	0.9970 ± 0.0007	0.3135	0.3150
37	PNL-3926 (4A.12)	Lead Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3159
38	PNL-3926 (4A.12)	Lead Reflector, 0.55 cm sepn.	4.306	1.0025 ± 0.0011	0.9997 ± 0.0007	0.3030	0.3044
39	PNL-3926 (4A.12)	Lead Reflector, 1.956 cm sepn.	4.306	1.0000 ± 0.0012	0.9985 ± 0.0007	0.2883	0.2930

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF ' (eV) MCNP4a	EALF ' (eV) KENO5a
40	PNL-3926 (4A.12)	Lead Reflector, 5.405 cm sepn.	4.306	0.9971 ± 0.0012	0.9946 ± 0.0007	0.2831	0.2854
41	PNL-2615 (4A.13)	Experiment 004/032 - no absorber	4.306	0.9925 ± 0.0012	0.9950 ± 0.0007	0.1155	0.1159
42	PNL-2615 (4A.13)	Experiment 030 - Zr plates	4.306	NC	0.9971 ± 0.0007	NC	0.1154
43	PNL-2615 (4A.13)	Experiment 013 - Steel plates	4.306	NC	0.9965 ± 0.0007	NC	0.1164
44	PNL-2615 (4A.13)	Experiment 014 - Steel plates	4.306	NC	0.9972 ± 0.0007	NC	0.1164
45	PNL-2615 (4A.13)	Exp. 009 1.05% Boron-Steel plates	4.306	0.9982 ± 0.0010	0.9981 ± 0.0007	0.1172	0.1162
46	PNL-2615 (4A.13)	Exp. 012 1.62% Boron-Steel plates	4.306	0.9996 ± 0.0012	0.9982 ± 0.0007	0.1161	0.1173
47	PNL-2615 (4A.13)	Exp. 031 - Boral plates	4.306	0.9994 ± 0.0012	0.9969 ± 0.0007	0.1165	0.1171
48	PNL-7167 (4A.14)	Experiment 214R - with flux trap	4.306	0.9991 ± 0.0011	0.9956 ± 0.0007	0.3722	0.3812
49	PNL-7167 (4A.14)	Experiment 214V3 - with flux trap	4.306	0.9969 ± 0.0011	0.9963 ± 0.0007	0.3742	0.3826
50	PNL-4267 (4A.15)	Case 173 - 0 ppm B	4.306	0.9974 ± 0.0012	NC	0.2893	NC
51	PNL-4267 (4A.15)	Case 177 - 2550 ppm B	4.306	1.0057 ± 0.0010	NC	0.5509	NC
52	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 21	20% Pu	1.0041 ± 0.0011	1.0046 ± 0.0006	0.9171	0.8868

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff}		EALF † (eV)	
				MCNP4a	KENO5a	MCNP4a	KENO5a
53	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 43	20% Pu	1.0058 ± 0.0012	1.0036 ± 0.0006	0.2968	0.2944
54	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 13	20% Pu	1.0083 ± 0.0011	0.9989 ± 0.0006	0.1665	0.1706
55	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 32	20% Pu	1.0079 ± 0.0011	0.9966 ± 0.0006	0.1139	0.1165
56	WCAP-3385 (4A.17)	Saxton Case 52 PuO2 0.52" pitch	6.6% Pu	0.9996 ± 0.0011	1.0005 ± 0.0006	0.8665	0.8417
57	WCAP-3385 (4A.17)	Saxton Case 52 U 0.52" pitch	5.74	1.0000 ± 0.0010	0.9956 ± 0.0007	0.4476	0.4580
58	WCAP-3385 (4A.17)	Saxton Case 56 PuO2 0.56" pitch	6.6% Pu	1.0036 ± 0.0011	1.0047 ± 0.0006	0.5289	0.5197
59	WCAP-3385 (4A.17)	Saxton Case 56 borated PuO2	6.6% Pu	1.0008 ± 0.0010	NC	0.6389	NC
60	WCAP-3385 (4A.17)	Saxton Case 56 U 0.56" pitch	5.74	0.9994 ± 0.0011	0.9967 ± 0.0007	0.2923	0.2954
61	WCAP-3385 (4A.17)	Saxton Case 79 PuO2 0.79" pitch	6.6% Pu	1.0063 ± 0.0011	1.0133 ± 0.0006	0.1520	0.1555
62	WCAP-3385 (4A.17)	Saxton Case 79 U 0.79" pitch	5.74	1.0039 ± 0.0011	1.0008 ± 0.0006	0.1036	0.1047

Notes: NC stands for not calculated.

† EALF is the energy of the average lethargy causing fission.

†† These experimental results appear to be statistical outliers (>3σ) suggesting the possibility of unusually large experimental error. Although they could justifiably be excluded, for conservatism, they were retained in determining the calculational basis.

Table 4A.2

COMPARISON OF MCNP4a AND KENO5a CALCULATED REACTIVITIES[†]
FOR VARIOUS ENRICHMENTS

Enrichment	Calculated $k_{eff} \pm 1\sigma$	
	MCNP4a	KENO5a
3.0	0.8465 ± 0.0011	0.8478 ± 0.0004
3.5	0.8820 ± 0.0011	0.8841 ± 0.0004
3.75	0.9019 ± 0.0011	0.8987 ± 0.0004
4.0	0.9132 ± 0.0010	0.9140 ± 0.0004
4.2	0.9276 ± 0.0011	0.9237 ± 0.0004
4.5	0.9400 ± 0.0011	0.9388 ± 0.0004

† Based on the GE 8x8R fuel assembly.

Table 4A.3

MCNP4a CALCULATED REACTIVITIES FOR CRITICAL EXPERIMENTS WITH NEUTRON ABSORBERS

Ref.	Experiment		Δk Worth of Absorber	MCNP4a Calculated k_{eff}	EALF[†] (eV)
4A.13	PNL-2615	Boral Sheet	0.0139	0.9994±0.0012	0.1165
4A.7	B&W-1484	Core XX	0.0165	1.0008±0.0011	0.1724
4A.13	PNL-2615	1.62% Boron-steel	0.0165	0.9996±0.0012	0.1161
4A.7	B&W-1484	Core XIX	0.0202	0.9961±0.0012	0.2103
4A.7	B&W-1484	Core XXI	0.0243	0.9994±0.0010	0.1544
4A.7	B&W-1484	Core XVII	0.0519	0.9962±0.0012	0.2083
4A.11	PNL-3602	Boral Sheet	0.0708	0.9941±0.0011	0.3135
4A.7	B&W-1484	Core XV	0.0786	0.9910±0.0011	0.2092
4A.7	B&W-1484	Core XVI	0.0845	0.9935±0.0010	0.1757
4A.7	B&W-1484	Core XIV	0.1575	0.9953±0.0011	0.2022
4A.7	B&W-1484	Core XIII	0.1738	1.0020±0.0011	0.1988
4A.14	PNL-7167	Expt 214R flux trap	0.1931	0.9991±0.0011	0.3722

[†]EALF is the energy of the average lethargy causing fission.

Table 4A.4

COMPARISON OF MCNP4a AND KENO5a
CALCULATED REACTIVITIES† FOR VARIOUS ^{10}B LOADINGS

^{10}B, g/cm^2	Calculated k_{eff} ± 1σ	
	MCNP4a	KENO5a
0.005	1.0381 ± 0.0012	1.0340 ± 0.0004
0.010	0.9960 ± 0.0010	0.9941 ± 0.0004
0.015	0.9727 ± 0.0009	0.9713 ± 0.0004
0.020	0.9541 ± 0.0012	0.9560 ± 0.0004
0.025	0.9433 ± 0.0011	0.9428 ± 0.0004
0.03	0.9325 ± 0.0011	0.9338 ± 0.0004
0.035	0.9234 ± 0.0011	0.9251 ± 0.0004
0.04	0.9173 ± 0.0011	0.9179 ± 0.0004

† Based on a 4.5% enriched GE 8x8R fuel assembly.

6.0 STRUCTURAL/SEISMIC CONSIDERATIONS

6.1 Introduction

This section considers the structural adequacy of the new maximum density spent fuel racks under all loadings postulated for normal, seismic, and accident conditions at MP-3. The proposed pool layout is shown in Figure 2.1, chapter 2.

As discussed in Chapter 1, the reracking of MP-3 involves the addition of fifteen new high density racks to the existing capacity. At the time of the original rack installation, the state-of-the-art limited the seismic evaluation to single rack 3-D simulations. As we discuss in this chapter, it is now possible to model the entire assemblage of new rack modules in one comprehensive simulation known as the 3-D Whole Pool Multi-Rack (WPMR) analysis. In order to maintain continuity with the previous analysis methods, both single rack and WPMR analyses have been performed to establish the structural margins of safety in the MP-3 racks.

The analyses undertaken to confirm the structural integrity of the racks are performed in compliance with the USNRC Standard Review Plan [6.1.1] and the OT Position Paper [6.1.2]. For each of the analyses, an abstract of the methodology, modeling assumptions, key results, and summary of parametric evaluations are presented. Delineation of the relevant criteria are discussed in the text associated with each analysis.

6.2 Overview of Rack Structural Analysis Methodology

The response of a free-standing rack module to seismic inputs is highly nonlinear and involves a complex combination of motions (sliding, rocking, twisting, and turning), resulting in impacts and friction effects. Some of the unique attributes of the rack dynamic behavior include a large fraction of the total structural mass in a confined rattling motion, friction support of rack pedestals against lateral motion, and large fluid coupling effects due to deep submergence and motion of closely spaced adjacent structures.

Table 4A.6

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH VARIOUS SOLUBLE BORON CONCENTRATIONS

Reference	Experiment	Boron Concentration, ppm	Calculated k_{eff}	
			MCNP4a	KENO5a
4A.15	PNL-4267	0	0.9974 ± 0.0012	-
4A.8	B&W-1645	886	0.9970 ± 0.0010	0.9924 ± 0.0006
4A.9	B&W-1810	1337	1.0023 ± 0.0010	-
4A.9	B&W-1810	1899	1.0060 ± 0.0009	-
4A.15	PNL-4267	2550	1.0057 ± 0.0010	-

Table 4A.7

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH MOX FUEL

Reference	Case[†]	MCNP4a		KENO5a	
		k_{eff}	EALF[††]	k_{eff}	EALF[††]
PNL-5803 [4A.16]	MOX Fuel - Exp. No. 21	1.0041±0.0011	0.9171	1.0046±0.0006	0.8868
	MOX Fuel - Exp. No. 43	1.0058±0.0012	0.2968	1.0036±0.0006	0.2944
	MOX Fuel - Exp. No. 13	1.0083±0.0011	0.1665	0.9989±0.0006	0.1706
	MOX Fuel - Exp. No. 32	1.0079±0.0011	0.1139	0.9966±0.0006	0.1165
WCAP-3385-54 [4A.17]	Saxton @ 0.52" pitch	0.9996±0.0011	0.8665	1.0005±0.0006	0.8417
	Saxton @ 0.56" pitch	1.0036±0.0011	0.5289	1.0047±0.0006	0.5197
	Saxton @ 0.56" pitch borated	1.0008±0.0010	0.6389	NC	NC
	Saxton @ 0.79" pitch	1.0063±0.0011	0.1520	1.0133±0.0006	0.1555

Note: NC stands for not calculated

† Arranged in order of increasing lattice spacing.

†† EALF is the energy of the average lethargy causing fission.

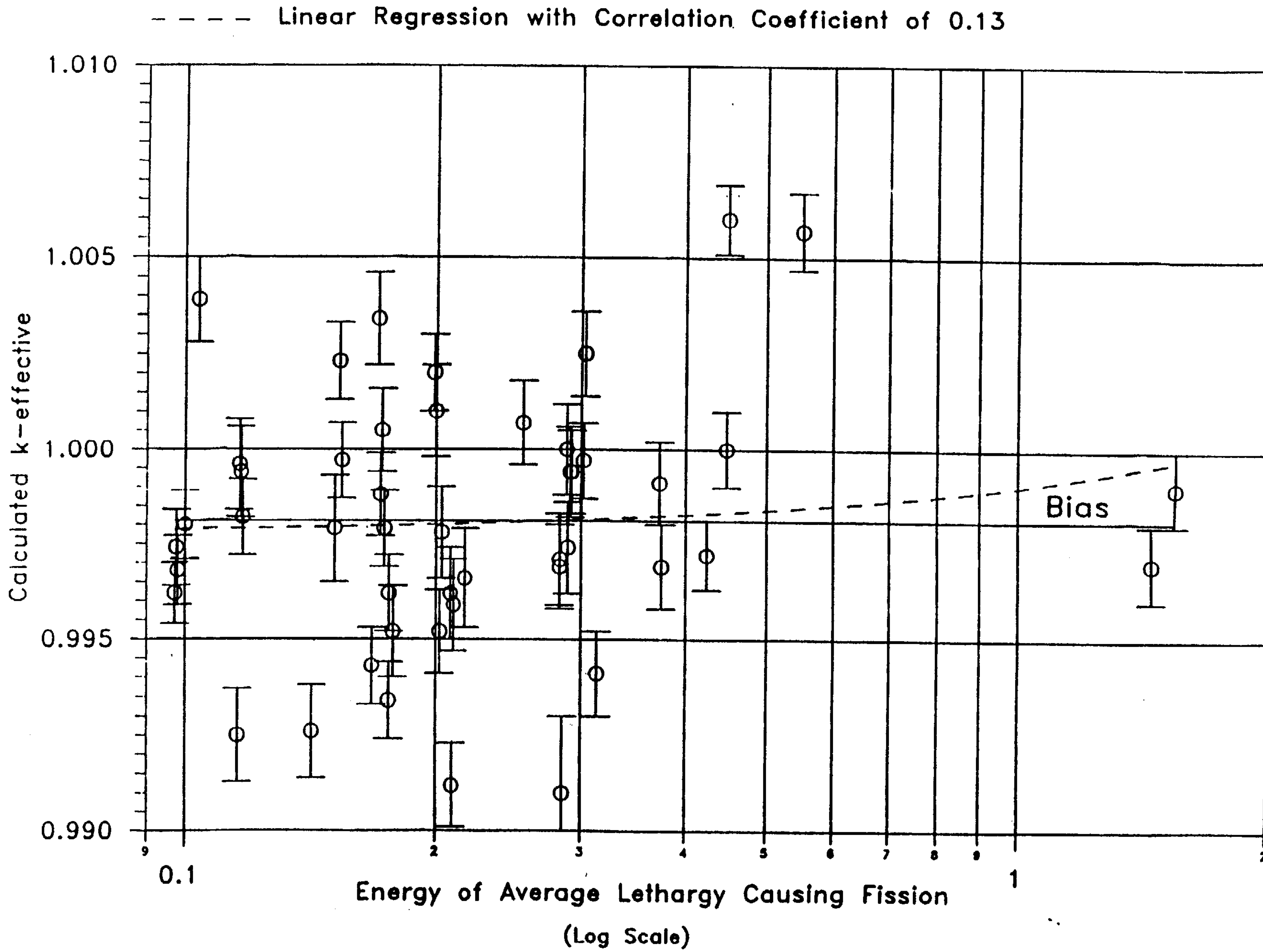


FIGURE 4A.1 MCNP CALCULATED k-eff VALUES for VARIOUS VALUES OF THE SPECTRAL INDEX

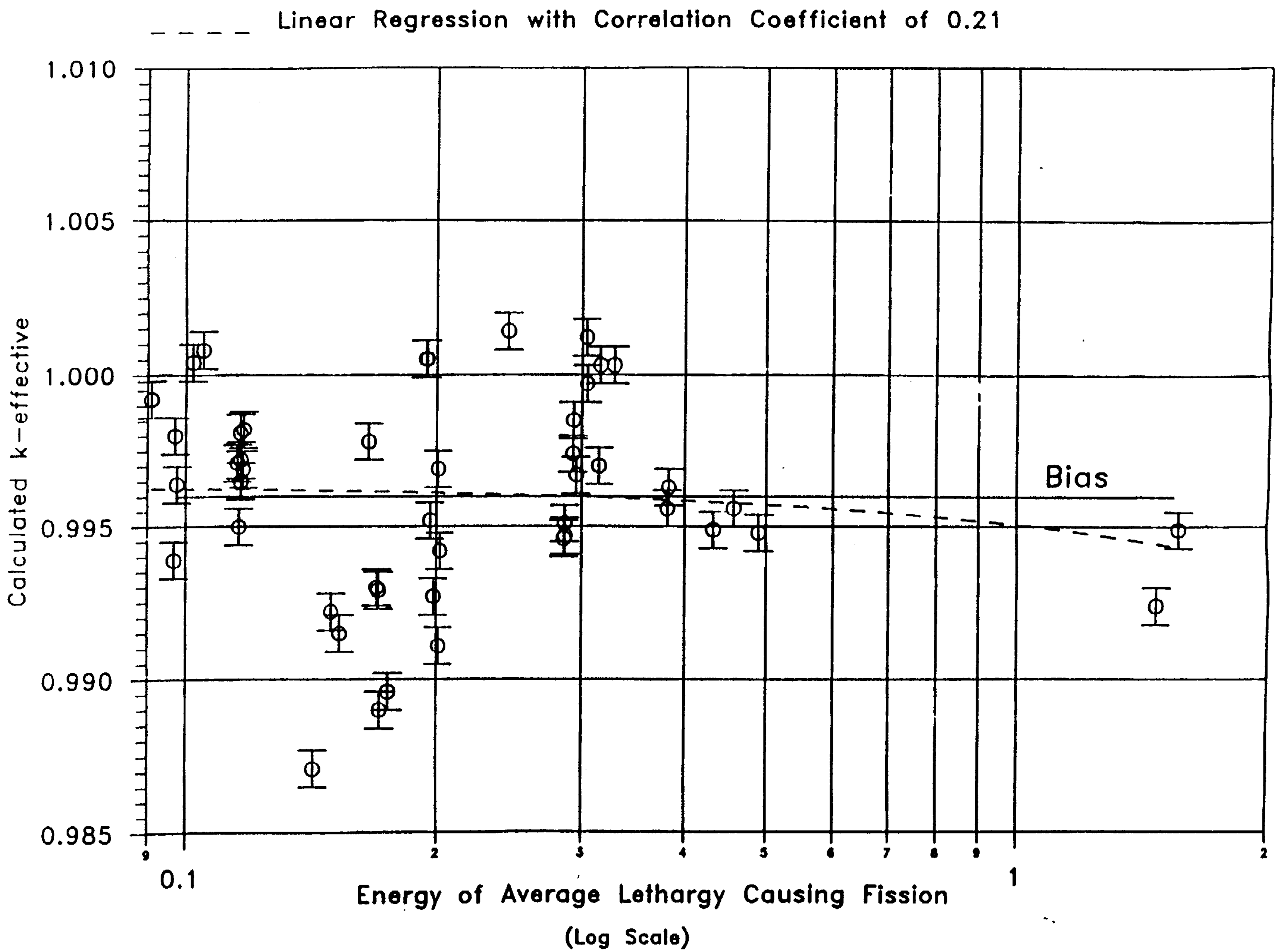


FIGURE 4A.2 KENO5a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

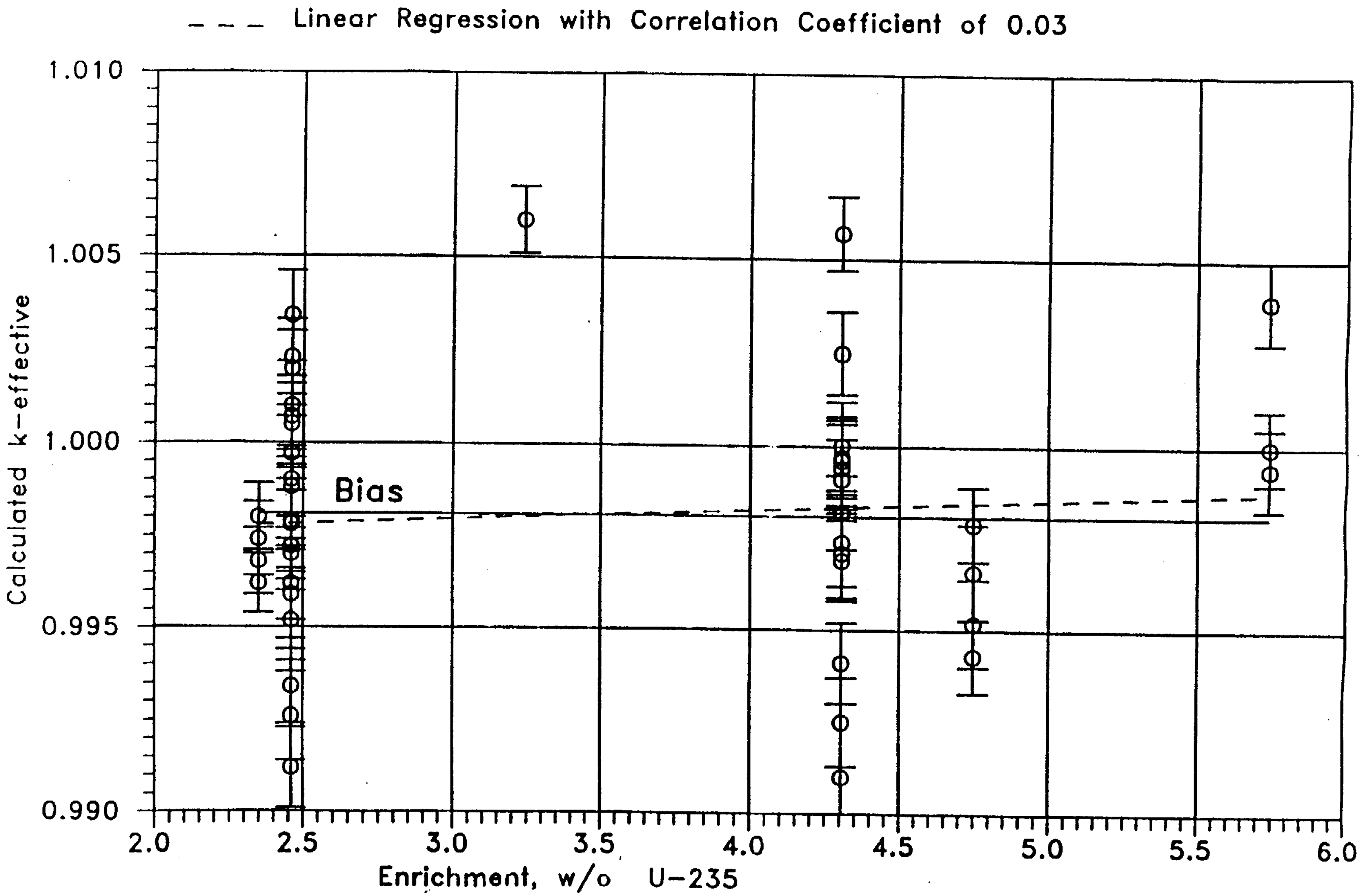


FIGURE 4A.3 MCNP CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

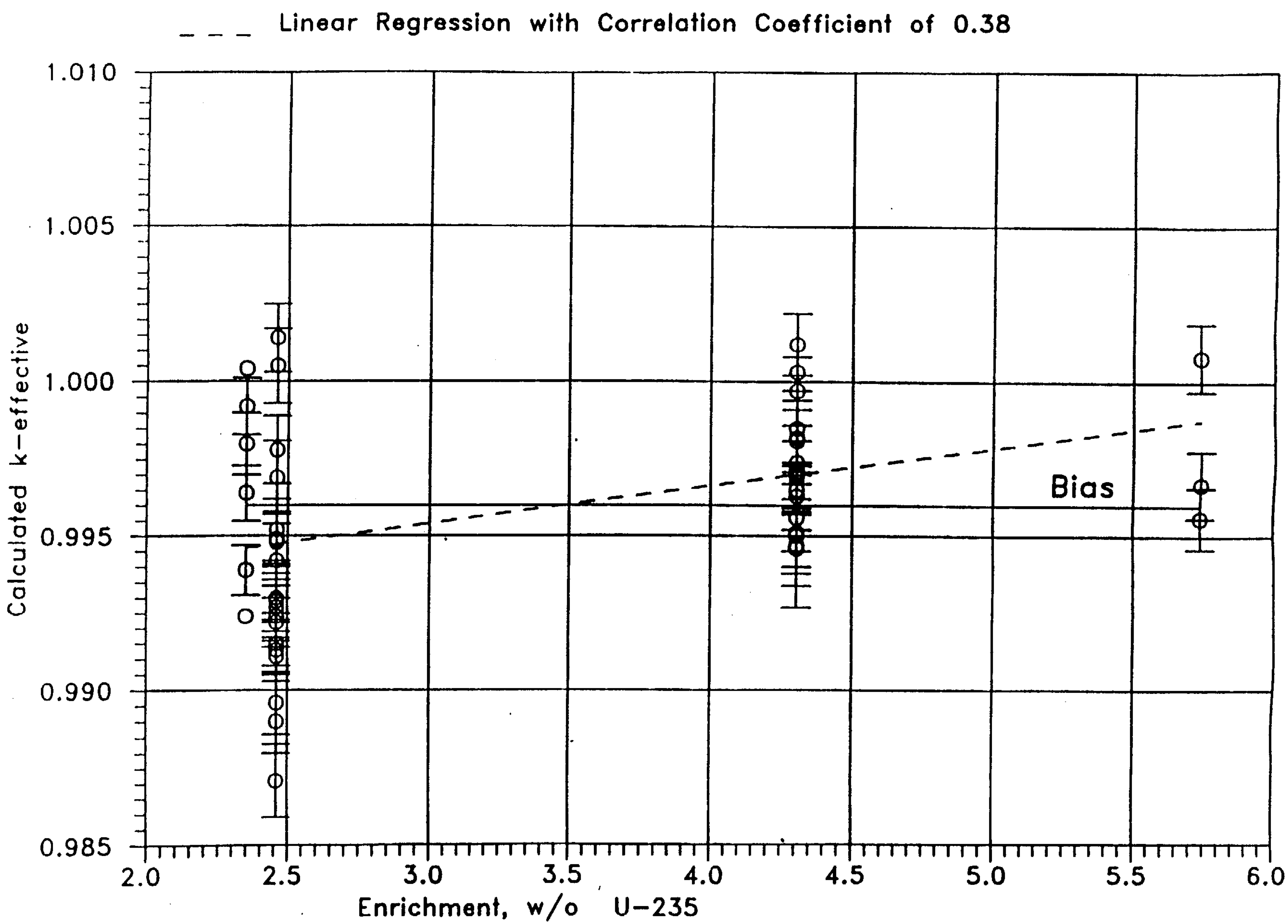


FIGURE 4A.4 KENO CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

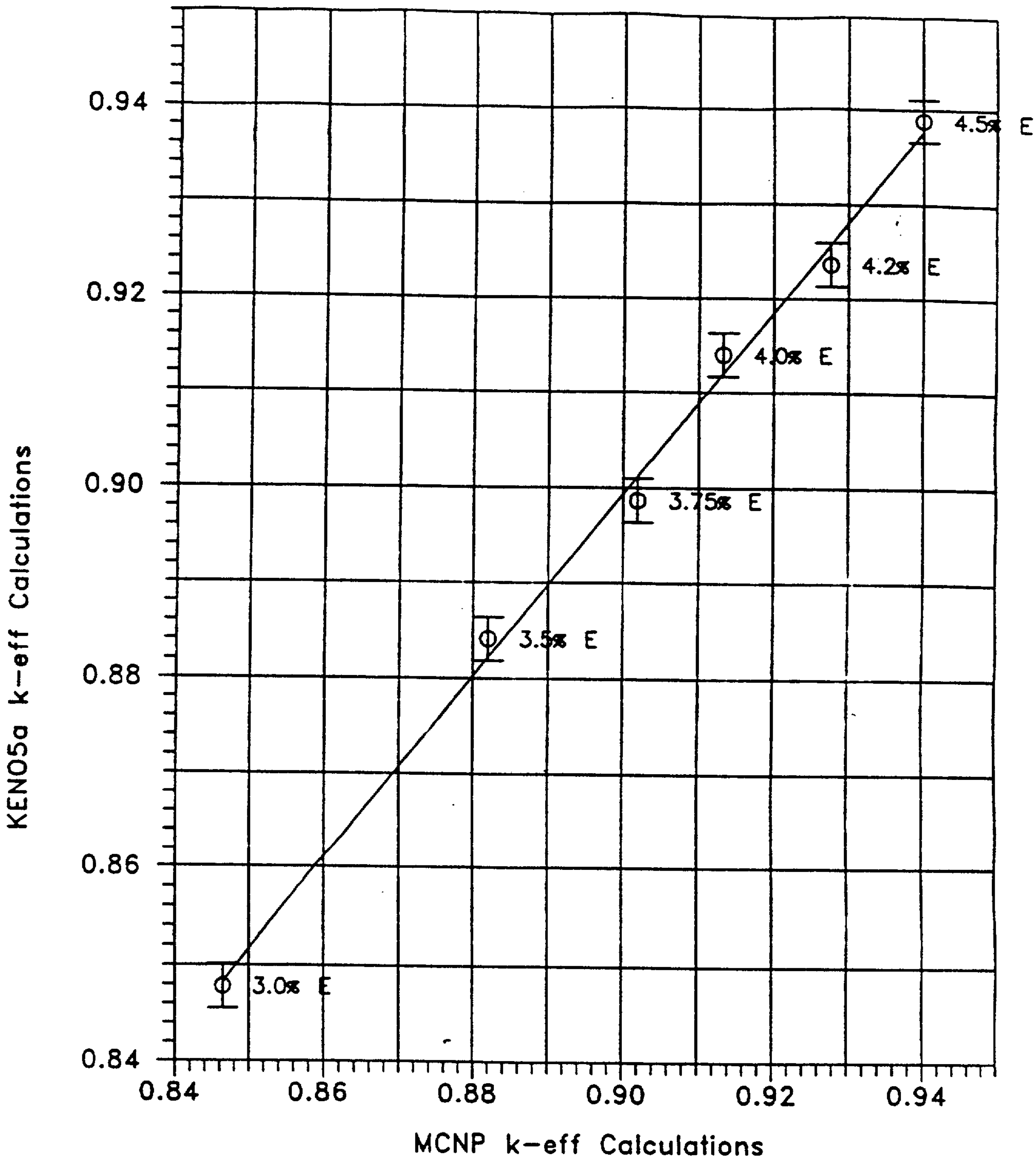


FIGURE 4A.5 COMPARISON OF MCNP AND KENO5A CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

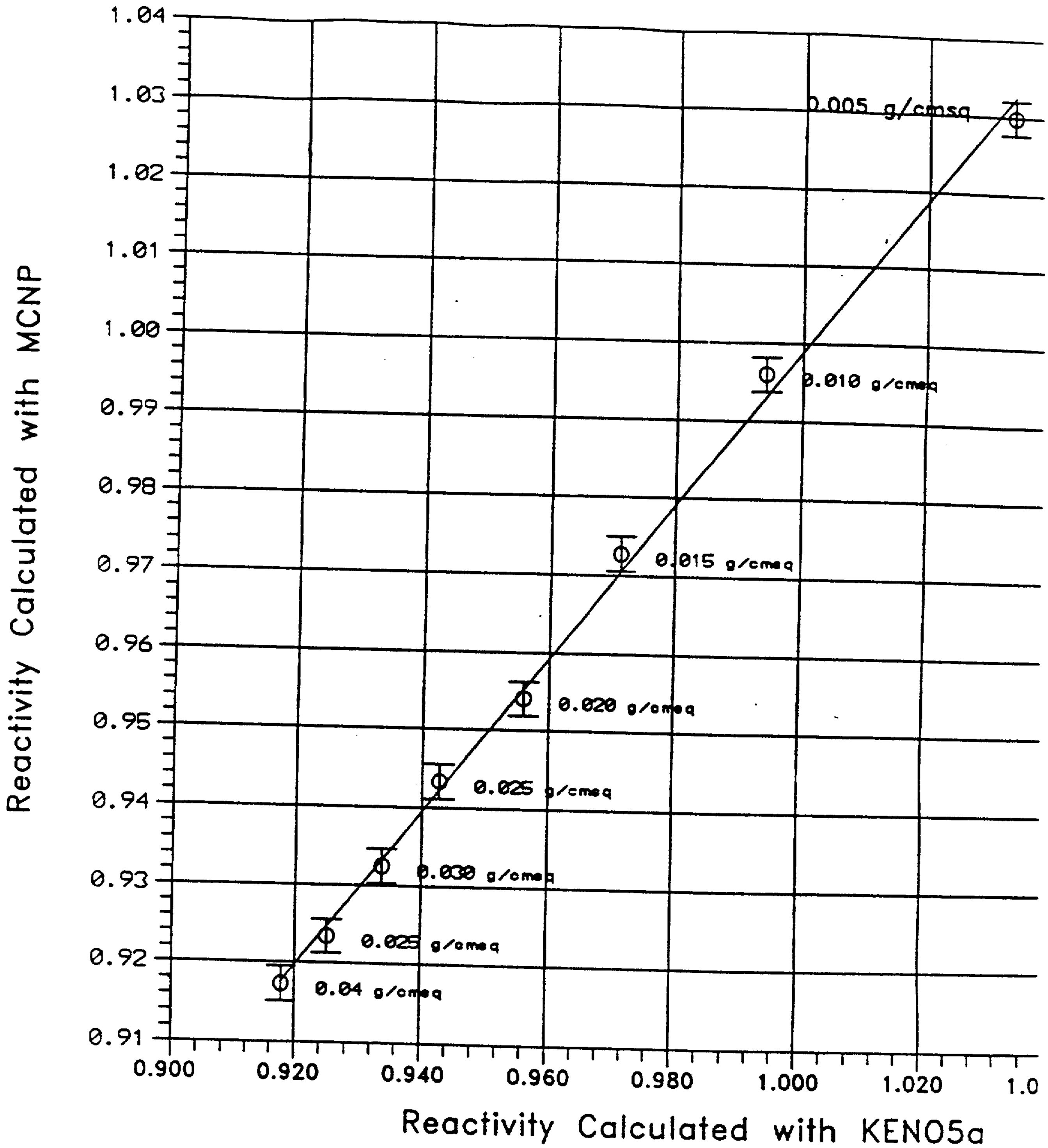


FIGURE 4A.6 COMPARISON OF MCNP AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES

5.0 THERMAL-HYDRAULIC CONSIDERATIONS

5.1 Introduction

This section provides a summary of the methods, models, analyses, and numerical results used to demonstrate compliance of the reracked Millstone Point Unit 3 (MP3) Spent Fuel Pool (SFP) and the Spent Fuel Pool Cooling and System (SFPCS) with the provisions of Section III of the USNRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications", (April 14, 1978). The methods used here are similar to methods of thermal-hydraulic analysis that have been used in other rerack licensing projects.

The thermal-hydraulic qualification analyses for the rack arrays may be broken down into the following categories:

i. Evaluation of the maximum decay heat load for the postulated discharge scenarios.

ii. Evaluation of the in-core hold times required to prevent exceeding the maximum temperature limit, as a function of component cooling water temperature.

iii. Evaluation of the postulated loss-of-forced cooling scenarios to establish that pool boiling will not occur.

iv. Determination of the maximum temperature difference between the pool local temperature and the bulk pool temperature at the instant when the bulk temperature reaches its maximum value, to establish that nucleate boiling at any location around the fuel is not possible with forced cooling available.

v. Evaluation of the maximum temperature difference between the fuel rod cladding temperature and the local pool water temperature to establish that departure from nucleate boiling (DNB) at any location around the fuel is not possible with forced cooling available.

A previous licensing submittal [5.1.1] has addressed items i through iii, above. The previous submittal is incorporated, by reference, into this document which addresses items iv and v. The following sections present the plant system description, analysis assumptions, a synopsis of the analysis methods employed, and final results.

5.2 System Description

The fuel pool cooling and purification system removes decay heat from spent fuel stored in the fuel pool and provides adequate cooling of water in the fuel pool. Two 100% capacity fuel pool cooling pumps and two 100% capacity fuel pool coolers are provided to ensure 100-percent redundant cooling capacity. This portion of the system is Seismic Category I and Safety Class 3. The spent fuel pool water flows from the fuel pool discharge through either of the two fuel pool cooling pumps and through the tubeside of either fuel pool cooler, and then returns to the fuel pool. Table 5.2.1 lists the performance characteristics of the fuel pool cooling system. Cooling for the fuel pool coolers is provided by the reactor plant component cooling water system.

Each pipe which enters the fuel pool has a vent hole drilled into the pipe to act as an anti-siphoning device or terminates at an elevation above these vent holes. These provisions prevent siphoning of the fuel pool water to less than 10 feet above the spent fuel. One pump and one cooler are sufficient to maintain the bulk pool temperatures to a maximum of 150°F for any long-term period. The bulk peak temperature of the spent fuel pool is limited to 200°F for structural qualification of the spent fuel pool.

Following a design basis accident with loss of power, the reactor plant component cooling water system is not available to cool the spent fuel pool coolers until approximately four hours after the accident. Power from the emergency generators is not immediately available due to loading considerations. Pool cooling will be reinitiated at this time.

Redundant safety grade fuel pool temperature indication is provided on the main control board. Redundant safety class 3 level instruments are located in the fuel pool and can be read from the control room. They are set to provide indication before the water level falls below 23 feet above the top of the fuel racks. Piping penetrations are at least 11 feet above the top of the spent fuel so that failure of inlets, outlets, or accident piping leaks cannot reduce the water below this level.

Normal makeup water to the spent fuel pool is the primary grade water system. Should primary grade water be unavailable, makeup water can be provided from the refueling water storage tank, a Seismic Category I source. Both of these systems connect to the spent fuel pool through the non-nuclear safety purification system. Water can also be provided from the hose station of the fire protection system near the spent fuel pool. As an additional safety feature, a Seismic Category I, Safety Class 3 flow path is provided from the service water system.

The fuel pool has redundant safety grade low level lights and temperature indicators provided in the main control room. Non-safety grade level indication is provided locally and high and low level alarms are provided both locally and in the control room.

Local temperature indicators are provided on each fuel pool cooler outlet. Fuel pool cooler outlet high temperature is alarmed locally. Fuel pool cooler outlet flow is indicated, and low flow alarmed, locally. Fuel pool cooler instrumentation is non-safety grade.

The fuel pool cooling pumps have control switches and indicating lights in the main control room. The discharges of all pumps have local pressure indicators. Upon high temperature at the pool, the plant will respond per procedural requirements. The cooling pumps can be operated manually either from the control room or the switchgear. The purification pumps are operated locally.

5.3 Discharge/Cooling Alignment Scenarios

The Millstone Unit 3 spent fuel pool is designed to meet the following post-reactor shutdown fuel discharge scenarios.

Case 1: Scheduled Full-Core Offload

One full core (193 assemblies) is off-loaded to the pool after one year of operation at full power.

Case 2: Unscheduled Full-core Offload

One full core (193 assemblies) is offloaded to the pool after a previous outage lasting for 10 days and followed by 36 days of operation at full power.

Case 3: Partial Core Discharge

This case is for a partial core discharge of up to 97 assemblies into the pool followed by loss of cooling for 4 hours. The temperature and decay heat loads in the pool at the start of loss of cooling correspond to the time at 600 hours after reactor shutdown. Component Cooling Water (CCP) temperature is assumed to be at an operating high temperature of 95°F.

In Case 1 and Case 2 discharge scenarios, it must be demonstrated that peak bulk pool temperatures do not exceed 150°F temperature limit when normal cooling is operational with CCP supplied to fuel pool heat exchanger. One fuel pool pump and one heat exchanger are assumed to be normally available for removing decay heat from the Millstone Unit 3 fuel pool for all scenarios. The two 100% capacity fuel pool cooling pumps and two 100% capacity fuel pool coolers are able to provide completely redundant cooling capacity.

The CCP system, following a design basis accident, is not available to cool the fuel pool for four hours. In the event of loss of pool cooling, it must be demonstrated that the bulk pool temperature shall not exceed 200°F during this four-hour post LOCA heat up of the pool.

5.4 Decay Heat Load, In-Core Hold Time, SFP Heat-Up Time

Section 4.0 of a previous licensing submittal [5.1.1] contains a description of the solution methodology used to evaluate the decay heat loads, in-core hold time requirements, and SFP heat-up times for the MP3 SFP and SFPCS. Please refer to that document for a discussion of the solution methodology for these evaluations. Note that for conservatism reference [5.1.1] assumed a higher end of cycle discharge size than assumed in Table 1.2 herein.

5.5 Local Pool Water Temperature

In this section, a summary of the methodology for evaluating the local pool water temperature is presented. A single conservative evaluation for a bounding amalgam of conditions is performed. The result of this single evaluation is a bounding temperature difference between the pool bulk temperature and the maximum local water temperature.

In order to determine an upper bound on the maximum local water temperature, a series of conservative assumptions are made. The most important of these assumptions are:

- For calculation of hydraulic resistance parameters, all racks are assumed to be Holtec designed Region 2 style racks. The lack of flux traps in this rack design minimizes the total flow area per stored assembly, thereby maximizing the hydraulic resistance and resultant temperatures.

- With a full core discharged into the racks farthest from the coolant water inlet, the remaining cells in the spent fuel pool are postulated to be occupied with previously discharged fuel.

- The hottest assemblies, located together in the pool, are assumed to be located in pedestal cells of the racks. These cells have a reduced water entrance area, caused by the pedestal blocking the baseplate hole, and a correspondingly increased hydraulic resistance.

- No downcomer flow is assumed to exist between the rack modules.

- All rack cells are conservatively assumed to be 50% blocked at the cell outlet to account for drop accidents resulting in damage to the upper end of the cells. This cell blockage is conservative, since structural evaluations have shown that only about 10% of the cell is blocked subsequent to the impact of dropped objects.

- The Westinghouse 17x17 Std. assembly, which is most resistive to axial fluid flow, is assumed to populate the entire storage region. Thus, the hydraulic resistance to heat transfer is maximized.

The inlet piping which returns cooled pool water from the SFPCS terminates above the level of the fuel racks. It is not apparent from heuristic reasoning alone that the cooled water delivered to the pool would not bypass the hot fuel racks and exit through the outlet piping. To demonstrate adequate cooling of hot fuel in the pool, it is therefore necessary to rigorously quantify the velocity field in the pool created by the interaction of buoyancy driven flows and water injection/egress. A Computational Fluid Dynamics (CFD) analysis for this demonstration is required. The objective of this study is to demonstrate that the principal thermal-hydraulic criteria of ensuring local subcooled conditions in the pool is met for all postulated fuel discharge/cooling alignment scenarios. The local thermal-hydraulic analysis is performed such that partial cell blockage and slight fuel assembly variations are bounded. An outline of the CFD approach is described in the following.

There are several significant geometric and thermal-hydraulic features of the MP3 SFP which need to be considered for a rigorous CFD analysis. From a fluid flow modeling standpoint, there are two regions to be considered. One region is the bulk pool region where the classical Navier-Stokes equations are solved with turbulence effects included. The other region is the heat generating fuel assemblies located in the spent fuel racks located near the bottom of the SFP. In this region, water flow is directed vertically upwards due to buoyancy forces through relatively small flow channels formed by the Westinghouse 17x17 fuel assembly rod arrays in each rack cell. This situation shall be modeled as a porous solid region in which fluid flow is governed by the classical Darcy's Law:

$$\frac{\partial P}{\partial X_i} = - \frac{\mu}{K(i)} V_i - C \rho |v_i| \frac{V_i}{2}$$

where $\partial P/\partial X_i$ is the pressure gradient, K(i), V_i and C are the corresponding permeability, velocity and inertial resistance parameters and μ is the fluid viscosity. The permeability and inertial resistance parameters for the rack cells loaded with Westinghouse 17×17 fuel were determined based on the friction factor correlations for the laminar flow conditions typically encountered due to the low buoyancy induced velocities and the small size of the flow channels.

The MP3 pool geometry requires an adequate portrayal of large scale and small scale features, spatially distributed heat sources in the spent fuel racks and water inlet/outlet configuration. Relatively cooler bulk pool water normally flows down between the fuel rack outline and pool wall liner clearance known as the downcomer. Near the bottom of the racks, the flow turns from a vertical to horizontal direction into the bottom plenum supplying cooling water to the rack cells. Heated water issuing out of the top of the racks mixes with the bulk pool water. An adequate modeling of these features on the CFD program involves meshing the large scale bulk pool region and small scale downcomer and bottom plenum regions with sufficient number of computational cells to capture the bulk and local features of the flow field.

The distributed heat sources in the spent fuel pool racks are modeled by identifying distinct heat generation zones considering full-core discharge, bounding peak effects, and presence of background decay heat from old discharges. Three heat generating zones were modeled. The first consists of background fuel from previous discharges, the remaining two zones consist of fuel from a bounding full-core-discharge scenario. The two full core discharge zones are differentiated by one zone with higher than average decay heat generation and the other with less than average decay heat generation. The background decay heat load is determined such that the total decay heat load in the pool is equal to the calculated decay heat load limit. This is a conservative model, since all of the fuel with higher than average decay heat is placed in a contiguous area. A uniformly distributed heat generation rate was applied throughout each distinct zone.

The CFD analysis was performed on the industry standard FLUENT [5.5.4] fluid flow and heat transfer modeling program. The FLUENT code enabled buoyancy flow and turbulence effects to be included in the CFD analysis. Turbulence effects are modeled by relating time-varying Reynolds Stresses to the mean bulk flow quantities with the following turbulence modeling options:

(i) k-ϵ Model
(ii) RNG k-ϵ Model
(iii) Reynolds Stress Model

The k-ϵ Model is considered most appropriate for the MP3 CFD analysis. The k-ϵ turbulence model is a time-tested, general purpose turbulence model. This model has been demonstrated to give good results for the majority of turbulent fluid flow phenomena. The Renormalization Group (RNG) and Reynolds Stress models are more advanced models that were developed for situations where the k-ϵ Model does not provide acceptable results, such as high viscosity flow and supersonic shock. The flow regime in the bulk fluid region is such that the k-ϵ Model will provide acceptable results.

Rigorous modeling of fluid flow problems requires a solution to the classical Navier-Stokes equations of fluid motion [5.5.1]. The governing equations (in modified form for turbulent flows with buoyancy effects included) are written as:

$$\frac{\partial \rho_o u_i}{\partial t} + \frac{\partial \rho_o \langle u'_i u'_j \rangle}{\partial x_i} = \frac{\partial}{\partial x_j}\left[\mu\left(\frac{\partial u_i}{\partial x_j} + \frac{\partial u_j}{\partial x_i}\right)\right]$$
$$- \frac{\partial P}{\partial x_i} - \rho_o \beta (T - T_o)\, g_i + \frac{\partial \rho_o \langle u'_i u'_j \rangle}{\partial x_j}$$

where u_i are the three time-averaged velocity components, $\rho\langle u'_i\, u'_j\rangle$ are time-averaged Reynolds stresses derived from the turbulence induced fluctuating velocity components u'_i, ρ_o is the fluid density at temperature T_o, β is the coefficient of thermal expansion, μ is the fluid viscosity, g_i are the components of gravitational acceleration and x_i are the Cartesian coordinate directions. The Reynolds stress tensor is expressed in terms of the mean flow quantities by defining a turbulent viscosity μ_t and a turbulent velocity scale k as shown below [5.5.2]:

$$\rho\langle u'_i u'_j\rangle = \frac{2}{3}\rho\, k\, \delta_{ij} - \mu_t \left[\frac{\partial u_i}{\partial x_j} + \frac{\partial u_j}{\partial x_i}\right]$$

The procedure to obtain the turbulent viscosity and velocity length scales involves a solution of two additional transport equations for kinetic energy (k) and rate of energy dissipation (ϵ). This methodology is known as the k-ϵ model for turbulent flows as described by Launder and Spalding [5.5.3].

Some of the major input values for this analysis are summarized in Table 5.5.1. An elevation view of the assembled CFD model is presented in Figure 5.5.1. Figures 5.5.2 and 5.5.3 present converged temperature contours and converged velocity vectors, respectively.

5.6 Fuel Rod Cladding Temperature

In this section. the method to calculate the temperature of the fuel rod cladding is presented. Similar to the local water temperature calculation methodology presented in the preceding section, this evaluation is performed for a single, bounding scenario. The maximum fuel cladding superheat above the local water temperature is calculated.

The maximum specific power of a fuel array q_A can be given by:

$$q_A = q\, F_{xy}$$

where:

F_{xy} = Radial peaking factor

q = Average fuel assembly specific power, Btu/hr

The peaking factors are given in Table 5.5.1. The maximum temperature rise of pool water in the most disadvantageously placed fuel assembly, defined as the one which is subject to the highest local pool water temperature, was computed for a bounding case. Having determined the maximum local water temperature in the pool, it is now possible to determine the maximum fuel cladding temperature. A fuel rod can produce F_z times the average heat emission rate over a small length, where F_z is the axial rod peaking factor. The axial heat distribution in a rod is generally a maximum in the central region, and tapers off at its two extremities. Thus, peak cladding heat flux over an infinitesimal area is given by the equation:

$$q_c = \frac{q\ F_{xy}\ F_z}{A_c}$$

where A_c is the total cladding external heat transfer area in the active fuel length region.

Within each fuel assembly sub-channel, water is continuously heated by the cladding as it moves axially upwards from bottom to top under laminar flow conditions. Rohsenow and Hartnett [5.6.1] report a Nusselt-number based heat transfer correlation for laminar flow in a heated channel. The film temperature driving force (ΔT_f) at the peak cladding flux location is calculated as follows:

$$h_f \frac{D_h}{Kw} = Nu$$

$$\Delta T_f = \frac{q_c}{h_f}$$

where, h_f is the water side film heat transfer coefficient, D_h is sub-channel hydraulic diameter, Kw is water thermal conductivity and Nu is the Nusselt number for laminar flow heat transfer.

In order to introduce some additional conservatism in the analysis, we assume that the fuel cladding has a crud deposit resistance R_c (equal to 0.0005 ht-ft^2-°F/Btu), which covers the entire surface. Thus, including the temperature drop across the crud resistance, the cladding to water local temperature difference (ΔT_c) is given by:

$$\Delta T_c = \Delta T_f + R_c\ q_c$$

5.7 Results

Section 5.0 of a previous licensing submittal [5.1.1] contains a summary presentation of the results of evaluations of the decay heat loads, in-core hold time requirements, and SFP heat-up times for the MP3 SFP and SFPCS. Please refer to that document for a discussion the results of these evaluations. A summary of the results of the local water and fuel cladding evaluation is presented below.

Consistent with our approach to make conservative assessments of temperature, the local water temperature calculations are performed for a pool with decay heat generation equal to the maximum calculated decay heat load limit. Thus, the local water temperature evaluation is a calculation of the temperature increment over the theoretical spatially uniform value due to local hot spots (due to the presence of a highly heat emissive fuel bundle).

The CFD study has analyzed a single bounding local thermal-hydraulic scenario. In this scenario, a bounding full-core discharge is considered in which the 193 assemblies are located in the pool, farthest from the cooled water inlet, while the balance of the rack cells are postulated to be occupied by fuel from old discharges.

The maximum temperature difference between the SFP bulk temperature and the peak local water temperature is conservatively calculated to be 41.67°F. The maximum temperature difference between the fuel cladding and the local water is calculated to be 36.31°F. Applying both calculated temperature differences to the bulk maximum normal operating pool temperature of 150°F [5.1.1] yields a conservatively bounding 191.67°F maximum local water temperature and a conservatively bounding 227.98°F peak cladding temperature. Both the maximum local water and fuel cladding temperatures are lower than the 239.45°F local boiling temperature on top of the racks. Thus, boiling does not occur anywhere within the MP3 SFP.

5.8 References

[5.1.1] "Licensing Report for Reclassification of Discharge in Millstone Point Unit 3 Spent Fuel Pool," Holtec Report HI-971843, Revision 2.

[5.5.1] Batchelor, G.K.. "An Introduction to Fluid Dynamics," Cambridge University Press, 1967.

[5.5.2] Hinze, J.O., "Turbulence," McGraw Hill Publishing Co., New York, NY, 1975.

[5.5.3] Launder, B.E., and Spalding, D.B., "Lectures in Mathematical Models of Turbulence", Academic Press, London. 1972.

[5.5.4] "QA Documentation and Validation of the FLUENT Version 4.32 CFD Analysis Program," Holtec Report HI-961444.

[5.6.1] Rohsenow, N.M., and Hartnett, J.P., "Handbook of Heat Transfer", McGraw Hill Book Company, New York, 1973.

Table 5.2.1

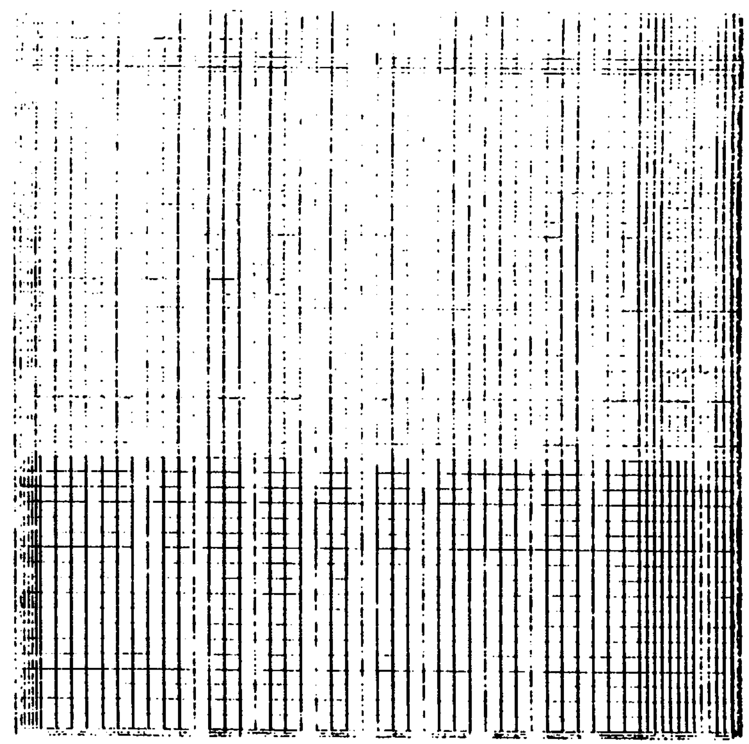
FUEL POOL COOLING AND PURIFICATION SYSTEM PRINCIPAL COMPONENT DESIGN CHARACTERISTICS

Fuel Pool Cooling Pumps	
Quantity	2
Capacity (gpm)	3,500
Head (ft)	115
Design pressure (psig)	200
Design temperature (°F)	200
Fuel Pool Heat Exchangers	
Quantity	2
Design heat load per exchanger (Btu/hr)	27.7×10^6
Reactor plant component cooling water flow per exchanger (gpm)	1,800
Reactor plant component cooling water inlet temperature (°F)	95
Reactor plant component cooling water outlet temperature (°F)	126
Fuel pool cooling flow (gpm)	3,500
Fuel pool water inlet temperature (°F)	140
Fuel pool water outlet temperature (°F)	125
Tubeside design pressure (psig)	150
Design temperature (°F)	200

TABLE 5.5.1

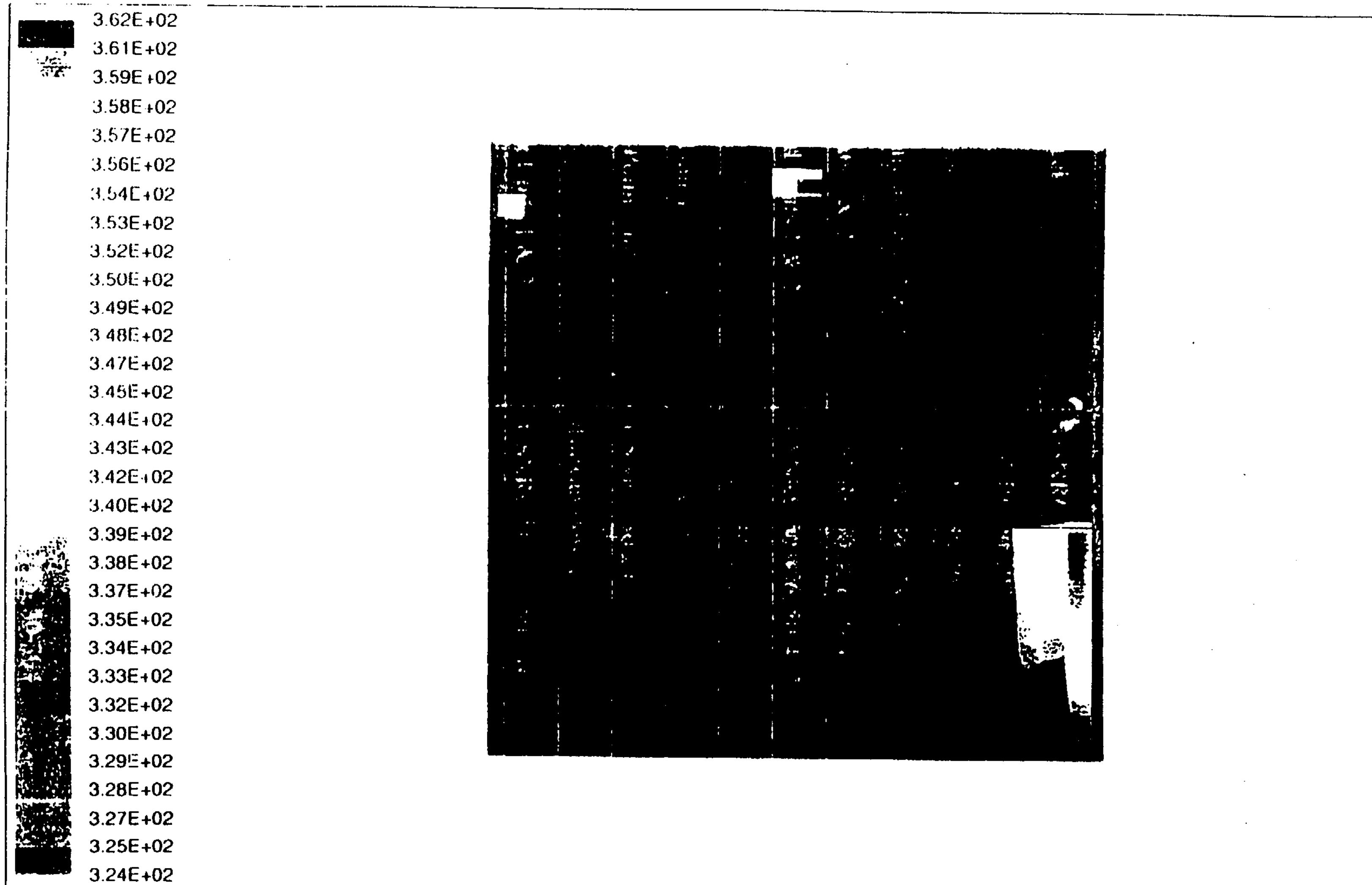
PRIMARY DATA FOR LOCAL TEMPERATURE EVALUATION

Fuel Rod Outer Diameter	0.374 in.
Rack Cell Inner Dimension	8.80 in.
Active Fuel Length	144 in.
SFPCS Water Flow Rate	3500 gpm
Fuel Radial Peaking Factor	1.70
Fuel Total Peaking Factor	2.60
SFP Length (North-South) Neglecting Southwest Area Opposite Cask Pit	355.82 in.
SFP Width (East-West)	452.41 in.
East Wall Minimum Rack-to-Wall Gap	4.17 in.
West Wall Minimum Rack-to-Wall Gap	6.31 in.
North Wall Minimum Rack-to-Wall Gap	3.17 in.
Minimum Rack-to-Floor (Bottom Plenum) Height	4.25 in.
Rack Cell Height (including baseplate)	170.0 in.
SFP Floor Liner Elevation	11 ft. & 3.25 in.
SFPCS Inlet Pipe Elevation	46 ft. & 4 in.
SFPCS Inlet Pipe Diameter	12 in. Sch. 40S
SFPCS Outlet Pipe Truncation Elevation	44 ft. & 5 in.
SFPCS Outlet Pipe Diameter	10 in. Sch. 40S
SFP Low Water Alarm Water Elevation	48 ft. & 11 in.



Y / Z / X	MILLSTONE POINT UNIT 3 TWO-DIMENSIONAL CFD MODEL Grid (67 X 57)	Mar 18 1998 Fluent 4.32 Fluent Inc.

FIGURE 5.5.1: Two-Dimensional Spent Fuel Pool Geometry Grid



	MILLSTONE POINT UNIT 3 TWO-DIMENSIONAL CFD MODEL Temperature (K) At Cell Centers Max = 3.618E+02 Min = 3.240E+02	Mar 18 1998 Fluent 4.32 Fluent Inc.

FIGURE 5.5.2: Two-Dimensional Spent Fuel Pool Converged Temperature Contours

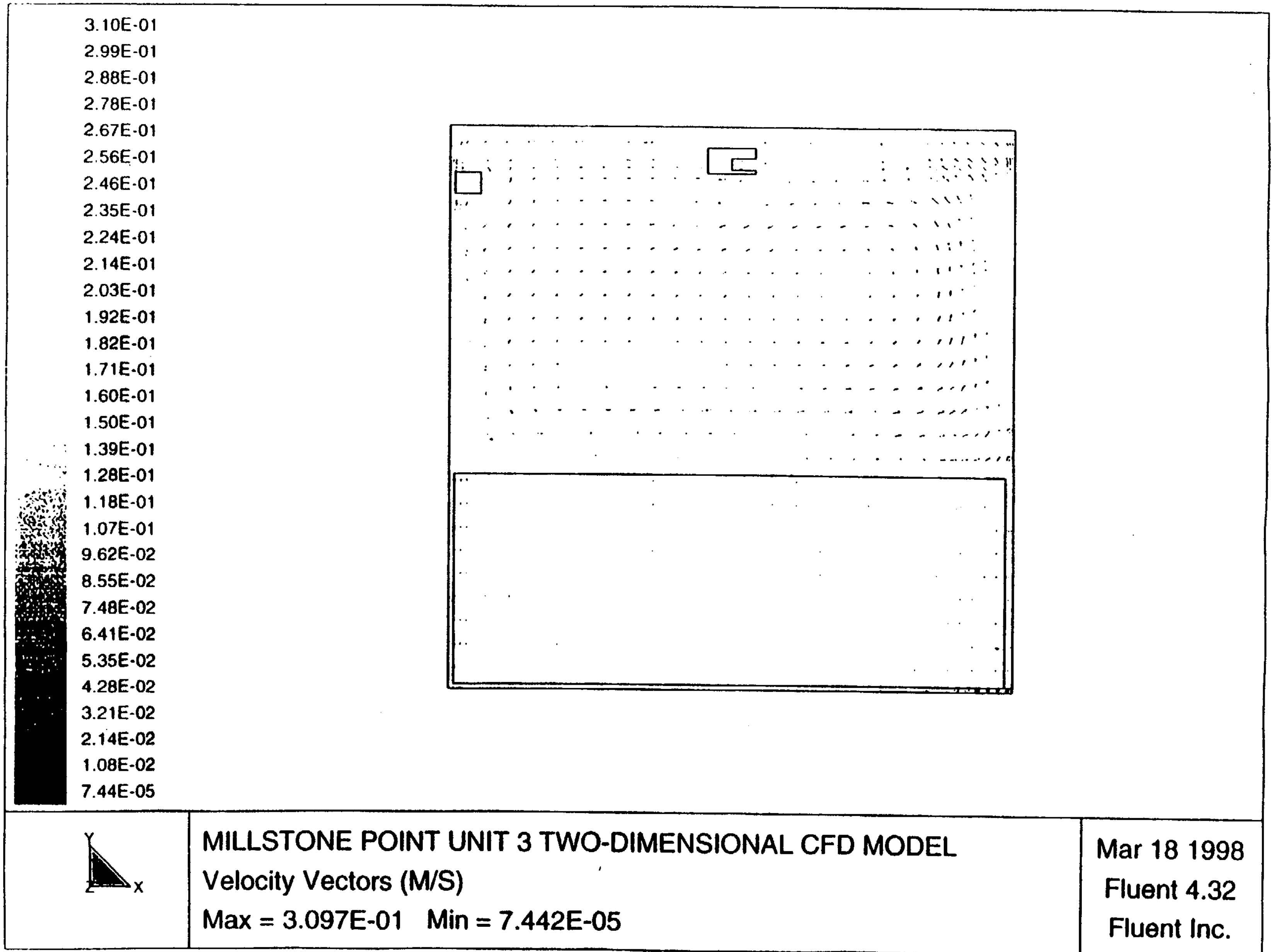


FIGURE 5.5.3: Two-Dimensional Spent Fuel Pool Converged Velocity Vectors

Linear methods, such as modal analysis and response spectrum techniques, cannot accurately simulate the structural response of such a highly nonlinear structure to seismic excitation. An accurate simulation is obtained only by direct integration of the nonlinear equations of motion with the three pool slab acceleration time-histories applied as the forcing functions acting simultaneously.

Both Whole Pool Multi-Rack (WPMR) and Single Rack analysis are used in this project to simulate the dynamic behavior of the complex storage rack structures. The following sections provide the basis for this selection and discussion on the development of the methodology.

6.2.1 Background of Analysis Methodology

Reliable assessment of the stress field and kinematic behavior of the rack modules calls for a conservative dynamic model incorporating all *key attributes* of the actual structure. This means that the model must feature the ability to execute the concurrent motion forms compatible with the free-standing installation of the modules.

The model must possess the capability to effect momentum transfers which occur due to rattling of fuel assemblies inside storage cells and the capability to simulate lift-off and subsequent impact of support pedestals with the pool liner (or bearing pad). The contribution of the water mass in the interstitial spaces around the rack modules and within the storage cells must be modeled in an accurate manner since erring in quantification of fluid coupling on either side of the actual value is no guarantee of conservatism.

The Coulomb friction coefficient at the pedestal-to-pool liner (or bearing pad) interface may lie in a rather wide range and a conservative value of friction cannot be prescribed *a priori*. In fact, a perusal of results of rack dynamic analyses in numerous dockets (Table 6.2.1) indicates that an upper bound value of the coefficient of friction often maximizes the computed rack displacements as well as the equivalent elastostatic stresses.

In short, there are a large number of parameters with potential influence on the rack kinematics. The comprehensive structural evaluation must deal with all of these without sacrificing conservatism.

The three-dimensional single rack dynamic model introduced by Holtec International personnel in the Enrico Fermi Unit 2 rack project (ca. 1980) and used in some 50 rerack projects since that time (Table 6.2.1) addresses most of the above mentioned array of parameters. The details of this methodology are also published in the permanent literature [6.2.1]. Despite the versatility of the 3-D seismic model, the accuracy of the single rack simulations has been suspect due to one key element; namely, hydrodynamic participation of water around the racks. During dynamic rack motion, hydraulic energy is either drawn from or added to the moving rack, modifying its submerged motion in a significant manner. Therefore, the dynamics of one rack affects the motion of all others in the pool.

However, Single Rack analysis is still a valuable tool to examine the behavior of a rack under different load conditions. It is used here as a first step in evaluating the racks. WPMR analysis builds upon the Single Rack model. The worst case loads and stresses that result from either of these two models are used to determine the structural adequacy of the racks.

The 3-D rack model dynamic simulation, involving one or more spent fuel racks, handles the array of variables as follows:

<u>Interface Coefficient of Friction</u> Parametric runs are made with upper bound and lower bound values of the coefficient of friction. The limiting values are based on experimental data which have been found to be bounded by the values 0.2 and 0.8. Simulations are also performed with the array of pedestals having randomly chosen coefficients of friction in a Gaussian distribution with a mean of 0.5 and lower and upper limits of 0.2 and 0.8, respectively. In the fuel rack simulations, the Coulomb friction interface between rack support pedestal and liner is simulated by piecewise linear (friction) elements. These elements function only when the pedestal is physically in contact with the pool liner.

Rack Beam Behavior Rack elasticity, relative to the rack base, is included in the model by introducing linear springs to represent the elastic bending action, twisting, and extensions.

Impact Phenomena Compression-only gap elements are used to provide for opening and closing of interfaces such as the pedestal-to-bearing pad interface, and the fuel assembly-to-cell wall interface. These interface gaps are modeled using nonlinear spring elements. The term "nonlinear spring" is a generic term used to denote the mathematical representation of the condition where a restoring force is not linearly proportional to displacement.

Fuel Loading Scenarios The fuel assemblies are conservatively assumed to rattle in unison which obviously exaggerates the contribution of impact against the cell wall.

Fluid Coupling Holtec International extended Fritz's classical two-body fluid coupling model to multiple bodies and utilized it to perform the first two-dimensional multi-rack analysis (Diablo Canyon, ca. 1987). Subsequently, laboratory experiments were conducted to validate the multi-rack fluid coupling theory. This technology was incorporated in the computer code DYNARACK [6.2.4] which handles simultaneous simulation of all racks in the pool as a Whole Pool Multi-Rack 3-D analysis. This development was first utilized in Chinshan, Oyster Creek, and Shearon Harris plants [6.2.1, 6.2.3] and, subsequently, in numerous other rerack projects. The WPMR analyses have corroborated the accuracy of the single rack 3-D solutions in predicting the maximum structural stresses, and also serve to improve predictions of rack kinematics.

For closely spaced racks, demonstration of kinematic compliance is verified by including all modules in one comprehensive simulation using a WPMR model. In WPMR analysis, all rack modules are modeled simultaneously and the coupling effect due to this multi-body motion is included in the analysis. Due to the superiority of this technique in predicting the dynamic behavior of closely spaced submerged storage racks, the Whole Pool Multi-Rack analysis methodology is used for this project.

6.3 Description of Racks

The storage capacity expansion of the MP-3 spent fuel pool features a two region arrangement. In the proposed scheme, five modules will store the most reactive fuel (up to 5 % w/o) without any burnup limitation. These racks will use a flux-trap design. The grouping of flux-trap racks is referred to as Region 1. The remaining new racks do not use flux-traps and are collectively referred to as Region 2. Region 2 racks have an enrichment/burnup limitation.

6.3.1 Fuel Weights

For the dynamic rack simulations, the dry fuel weight is conservatively taken to be 1700 lbs. The actual fuel assembly weight is approximately 1482 lbs. The higher fuel weight value of 1700 lbs is used to account for rod control cluster assemblies (RCCAs) being stored along with fuel assemblies. Therefore, the analyses conservatively consider an RCCA to be stored along with an assembly at every location.

6.4 Synthetic Time-Histories

The synthetic time-histories in three orthogonal directions (N-S, E-W, and vertical) are generated in accordance with the provisions of SRP 3.7.1 [6.4.1]. In order to prepare an acceptable set of acceleration time-histories, Holtec International's proprietary code GENEQ [6.4.2] is utilized.

A preferred criterion for the synthetic time-histories in SRP 3.7.1 calls for both the response spectrum and the power spectral density corresponding to the generated acceleration time-history to envelope their target (design basis) counterparts with only finite enveloping infractions. The time-histories for the pools have been generated to satisfy this preferred (and more rigorous) criterion. The seismic files also satisfy the requirements of statistical independence mandated by SRP 3.7.1. Figures 6.4.1 through 6.4.3 and 6.4.4 through 6.4.6 provide plots of the time-history accelerograms which were generated over a 20 second duration for SSE and OBE events, respectively. These artificial time-histories are used in all non-linear dynamic simulations of the racks.

Results of the correlation function of the three time-histories are given in Table 6.4.1. Absolute values of the correlation coefficients are shown to be less than 0.15, indicating that the desired statistical independence of the three data sets has been met.

6.5 22-DOF Nonlinear Rack Model for Dynamic Analysis

6.5.1 General Remarks

The single rack 3-D model of the MP-3 racks has been prepared with due consideration of the following characteristics, which are typical of high-density modules designed by Holtec International.

i. As a continuous structure, the rack possesses an infinite number of degrees-of-freedom, of which the cantilever beam type modes are most pronounced under seismic excitation if the rack is of the honeycomb construction genré. (The MP-3 racks, like all prior Holtec designs, are of the honeycomb type.)

ii. The fuel assemblies are "nimble" structures with a relatively low beam mode fundamental frequency.

iii. The interstitial gap between the storage cells and the stored fuel assemblies leads to a rattling condition in the storage cells during a seismic event.

iv. The lateral motion of the rack due to seismic input is resisted by the pedestal-to-pool slab interfacial friction and is abetted or retarded by the fluid coupling forces produced by the proximity of the rack to other structures. (The fluid coupling forces are distinct from the nonconservative forces such as fluid "drag" which are, by NRC regulations, excluded from the analysis). The construction of a 3-D single rack dynamic model consists of modeling the rack as a multi-degree-of-freedom structure such a manner that the selected DOFs capture all macro-motion modes of the rack. such as twisting, overturning, lift-off, sliding, flexing, and combinations thereof. Particular attention must be paid to incorporating the potential for the friction-resisted sliding of the rack on the liner, lift-off and subsequent impact of the pedestals on the slab. collision of the rack with adjacent structures, and most important. rattling of the fuel in the storage cells. The dynamic model must also provide for the ability to simulate the scenarios of partially loaded racks with arbitrary loading patterns.

As the name implies, the single rack (SR) dynamic model is a 3-D structural model for *one* rack in the pool. The rack selected for the SR analysis in this project is the one with the most mass, or most non-square cross section (i.e., aspect ratio). The dynamic model of this rack, i.e., its structural stiffness characteristics, rattling effect of the stored fuel, etc., can be prepared with extreme diligence in the manner described in the following, resulting in an excellent articulation of the rack structure. Even the fluid coupling effects between the fuel assemblies and the storage cell can be modeled with acceptable accuracy [6.5.2]. If the rack is adjacent to a wall, the fluid coupling effects between the rack and the wall can also be set down deterministically because the wall is a fixed structure. Such a definitive situation does not exist, however, when the neighboring structure to the subject rack is another free-standing rack. During a seismic event, the subject rack and the neighboring rack will both undergo dynamic motions which will be governed by the interaction among the inertia, fluid, friction, and rattling forces for each rack. The fluid coupling forces between two racks, however, depend on their *relative* motions. Because the motion of the neighboring rack is undefined, it is not possible to characterize the hydrodynamic forces arising from the fluid coupling between the neighboring rack and the subject rack. This inability to accurately model the inter-rack fluid coupling effects is a central limitation in the single rack analysis.

To overcome this limitation intrinsic to the single rack solutions, an artificial boundary condition, referred to as the "out-of-phase" assumption, has been historically made to bound the problem.

In the opposed-phase motion assumption, it is assumed that *all* racks adjacent to the subject rack are vibrating 180° out-of-phase, resulting in a plane of symmetry between the subject rack and the neighboring rack across which water will not flow. Thus, the subject rack is essentially surrounded by a fictitious box with walls that are midway to the adjacent racks. Impact with the adjacent rack is assumed to have occurred if the subject rack contacts the "box wall".

In summary, in the opposed-phase motion analysis the analyst makes the election that the adjacent racks are moving at 180° out-of-phase from the subject rack at all times during the seismic event. This is an artificial technical construct, albeit one that is known to predict rack-to-rack impact conservatively.

Therefore, to maintain consistency with past analyses, an array of single rack 3-D simulations were carried out, principally to compare the results (viz., rack-to-rack impact, maximum primary stress levels, pedestal loads, etc.) with the more definitive WPMR analysis. The description below provides the essentials of the 22 DOF model for a single rack. This model is used in both 3-D single rack simulations and as the building block for the more complicated WPMR analyses, described later in this chapter.

The dynamic modeling of the rack structure is prepared with special consideration of all nonlinearities and parametric variations. Particulars of modeling details and assumptions for the rack analysis are given in the following

a. The fuel rack structure motion is captured by modeling the rack as a 12 degree-of-freedom structure. Movement of the rack cross-section at any height is described by six degrees-of-freedom of the rack base and six degrees-of-freedom at the rack top. In this manner, the response of the module, relative to the baseplate, is captured in the dynamic analyses once suitable springs are introduced to couple the rack degrees-of-freedom and simulate rack stiffness.

b. Rattling fuel assemblies within the rack are modeled by five lumped masses located at H, .75H, .5H, .25H, and at the rack base (H is the rack height measured above the baseplate). Each lumped fuel mass has two horizontal displacement degrees-of-freedom. Vertical motion of the fuel assembly mass is assumed equal to rack vertical motion at the baseplate level. The centroid of each fuel assembly mass can be located off-center, relative to the rack structure centroid at that level, to simulate a partially loaded rack.

c. Seismic motion of a fuel rack is characterized by random rattling of fuel assemblies in their individual storage locations. All fuel assemblies are assumed to move in-phase within a rack. This exaggerates computed dynamic loading on the rack structure and, therefore, yields conservative results.

d. Fluid coupling between rack and fuel assemblies, and between rack and wall, is simulated by appropriate inertial coupling in the system kinetic energy. Inclusion of these effects uses the methods of [6.5.2, 6.5.3] for rack/assembly coupling and for rack-to-rack coupling.

e. Fluid damping and form drag are conservatively neglected.

f. Sloshing is found to be negligible at the top of the rack and is, therefore, neglected in the analysis of the rack.

g. Potential impacts between the cell walls of the new racks and the contained fuel assemblies are accounted for by appropriate compression-only gap elements between masses involved. The possible incidence of rack-to-wall or rack-to-rack impact is simulated by gap elements at the top and bottom of the rack in two horizontal directions. Bottom gap elements are located at the baseplate elevation. The initial gaps reflect the presence of baseplate extensions, and the rack stiffnesses are chosen to simulate local structural detail.

h. Pedestals are modeled by gap elements in the vertical direction and as "rigid links" for transferring horizontal stress. Each pedestal support is linked to the pool liner (or bearing pad) by two friction springs. The spring rate for the friction springs includes any lateral elasticity of the stub pedestals. Local pedestal vertical spring stiffness accounts for floor elasticity and for local rack elasticity just above the pedestal.

i. Rattling of fuel assemblies inside the storage locations causes the gap between fuel assemblies and cell wall to change from a maximum of twice the nominal gap to a theoretical zero gap. Fluid coupling coefficients are based on the nominal gap in order to provide a conservative measure of fluid resistance to gap closure.

j. The model for the rack is considered supported, at the base level, on four pedestals modeled as non-linear compression only gap spring elements and eight piecewise linear friction spring elements; these elements are properly located with respect to the centerline of the rack beam, and allow for arbitrary rocking and sliding motions.

6.5.2 Element Details

Figure 6.5.1 shows a schematic of the dynamic model of a single rack. The schematic depicts many of the characteristics of the model including all of the degrees-of-freedom and some of the spring restraint elements.

Table 6.5.1 provides a complete listing of each of the 22 degrees-of-freedom for a rack model. Six transitional and six rotational degrees-of-freedom (three of each type on each end) describe the motion of the rack structure. Rattling fuel mass motions (shown at nodes 1^*, 2^*, 3^*, 4^*, and 5^* in Figure 6.5.1) are described by ten horizontal transitional degrees-of-freedom (two at each of the five fuel masses). The vertical fuel mass motion is assumed (and modeled) to be the same as that of the rack baseplate.

Figure 6.5.2 depicts the fuel to rack impact springs (used to develop potential impact loads between the fuel assembly mass and rack cell inner walls) in a schematic isometric. Only one of the five fuel masses is shown in this figure. Four compression only springs, acting in the horizontal direction, are provided at each fuel mass.

Figure 6.5.3 provides a 2-D schematic elevation of the storage rack model. discussed in more detail in Section 6.5.3. This view shows the vertical location of the five storage masses and some of the support pedestal spring members.

Figure 6.5.4 shows the modeling technique and degrees-of-freedom associated with rack elasticity. In each bending plane a shear and bending spring simulate elastic effects [6.5.4]. Linear elastic springs coupling rack vertical and torsional degrees-of-freedom are also included in the model.

Figure 6.5.5 depicts the inter-rack impact springs (used to develop potential impact loads between racks or between rack and wall). The approximate spring contact location at rack top and bottom and the numbering of each impact spring used in the model are shown in Figure 6.8.1 and Figure 6.8.2.

6.5.3 Fluid Coupling Effect

In its simplest form, the so-called "fluid coupling effect" [6.5.2, 6.5.3] can be explained by considering the proximate motion of two bodies under water. If one body (mass m_1) vibrates adjacent to a second body (mass m_2), and both bodies are submerged in frictionless fluid, then Newton's equations of motion for the two bodies are:

$$(m_1 + M_{11})\ddot{X}_1 + M_{12}\ddot{X}_2 = \text{applied forces on mass } m_1 + O(X_1^2)$$

$$M_{21}\ddot{X}_1 + (m_2 + M_{22})\ddot{X}_2 = \text{applied forces on mass } m_2 + O(X_2^2)$$

$\ddot{X}_1$, and $\ddot{X}_2$ denote absolute accelerations of masses m_1 and m_2, respectively, and the notation $O(X^2)$ denotes nonlinear terms.

M_{11}, M_{12}, M_{21}, and M_{22} are fluid coupling coefficients which depend on body shape, relative disposition, etc. Fritz [6.5.3] gives data for M_{ij} for various body shapes and arrangements. The fluid adds mass to the body (M_{11} to mass m_1), and an inertial force proportional to acceleration of the adjacent body (mass m_2). Thus. acceleration of one body affects the force field on another. This force field is a function of inter-body gap. reaching large values for small gaps. Lateral motion of a fuel assembly inside a storage location encounters this effect. For example. fluid coupling behavior will be experienced between nodes 2 and 2* in Figure 6.5.1. The rack analysis also contains inertial fluid coupling terms which model the effect of fluid in the gaps between adjacent racks.

Terms modeling the effects of fluid flowing between adjacent racks in a single rack analysis suffer from the inaccuracies described earlier. These terms are computed assuming that all racks adjacent to the rack being analyzed are vibrating in-phase or 180° out of phase. The WPMR analyses do not require any assumptions with regard to phase.

6.5.4 Stiffness Element Details

Table 6.5.2 lists all spring elements used in the 3-D 22-DOF single rack model. It helps to explain the stiffness details. In the table, the following coordinate system applies:

x = Horizontal axis along plant North

y = Horizontal axis along plant West

z = Vertical axis upward from the rack base

If the simulation model is restricted to two dimensions (one horizontal motion plus one vertical motion. for example). for the purposes of model clarification only, then Figure 6.5.3 describes the configuration. This simpler model is used to elaborate on the various stiffness modeling elements.

Type 3 gap elements modeling impacts between fuel assemblies and racks have local stiffness K_i in Figure 6.5.3. In Table 6.5.2, for example, type 3 gap elements 5 through 8 act on the rattling fuel mass at the rack top. Support pedestal spring rates K_S are modeled by type 3 gap elements 1 through 4. as listed in Table 6.5.2. Local compliance of the concrete floor is included in K_S. The type 2 friction elements listed in Table 6.5.2 are shown in Figure 6.5.3 as K_f. The spring elements depicted in Figure 6.5.4 represent type 1 elements.

Friction at support/liner interface is modeled by the piecewise linear friction springs with suitably large stiffness K_f up to the limiting lateral load μN, where N is the current compression load at the interface between support and liner. At every time-step during transient analysis, the current value of N (either zero if the pedestal has lifted off the liner, or a compressive finite value) is computed.

The gap element K_S, modeling the effective compression stiffness of the structure in the vicinity of the support, includes stiffness of the pedestal, local stiffness of the underlying pool slab, and local stiffness of the rack cellular structure above the pedestal.

The previous discussion is limited to a 2-D model solely for simplicity. Actual analyses incorporate 3-D motions and include all stiffness elements listed in Table 6.5.2.

6.6 Whole Pool Multi-Rack Methodology

6.6.1 General Remarks

The single rack 3-D (22-DOF) models for the new racks outlined in the preceding subsection are used as a first step to evaluate the structural integrity and physical stability of the rack modules. However, prescribing the motion of the racks adjacent to the module being analyzed is an assumption in the single rack simulations which cannot be defended on the grounds of conservatism. For closely spaced racks, demonstration of the kinematic compliance is further verified by including all modules in one comprehensive simulation using a Whole Pool Multi-Rack (WPMR) model. The WPMR analysis builds on the Single Rack model by simultaneously modeling all racks; a coupling effect results due to the multi-body motion.

Recognizing that the analysis work effort must deal with both stress and displacement criteria, the sequence of model development and analysis steps that are undertaken are summarized in the following:

a. Prepare 3-D dynamic models suitable for a time-history analysis of the new maximum density racks. These models include the assemblage of all new rack modules in the pool. Include all fluid coupling interactions and mechanical coupling appropriate to performing an accurate non-linear simulation. This 3-D simulation is referred to as a Whole Pool Multi-Rack model.

b. Perform 3-D dynamic analyses on various physical conditions (such as coefficient of friction and extent of cells containing fuel assemblies). Archive appropriate displacement and load outputs from the dynamic model for post-processing.

c. Perform stress analysis of high stress areas for the limiting case of all the rack dynamic analyses. Demonstrate compliance with ASME Code Section III, Subsection NF limits on stress and displacement.

6.6.2 Multi-Body Fluid Coupling

During the seismic event, all racks in the pool are subject to the input excitation simultaneously. The motion of each free-standing module would be autonomous and independent of others as long as they do not impact each other and no water is present in the pool. While the scenario of inter-rack impact is not a common occurrence and depends on rack spacing, the effect of water - the so-called fluid coupling effect - is a universal factor. As noted in Ref. [6.5.2, 6.5.3], the fluid forces can reach rather large values in closely spaced rack geometries. It is, therefore, essential that the contribution of the fluid forces be included in a comprehensive manner. This is possible only if all racks in the pool are *allowed* to execute 3-D motion in the mathematical model. For this reason, single rack or even multi-rack models involving only a portion of the racks in the pool, are inherently inaccurate. The Whole Pool Multi-Rack model removes this intrinsic limitation of the rack dynamic models by simulating the 3-D motion of all modules simultaneously. The fluid coupling effect, therefore, encompasses interaction between *every* set of racks in the pool, i.e., the motion of one rack produces fluid forces on all other racks and on the pool walls. Stated more formally, both near-field and far-field fluid coupling effects are included in the analysis.

The derivation of the fluid coupling matrix [6.6.2] relies on the classical inviscid fluid mechanics principles, namely the principle of continuity and Kelvin's recirculation theorem. While the derivation of the fluid coupling matrix is based on no artificial construct, it has been nevertheless verified by an extensive set of shake table experiments [6.6.2].

6.6.3 Coefficients of Friction

To eliminate the last significant element of uncertainty in rack dynamic analyses, multiple simulations are performed to adjust the friction coefficient ascribed to the support pedestal/pool bearing pad interface. These friction coefficients are chosen consistent with the two bounding extremes from Rabinowicz's data [6.5.1]. Simulations are also performed by imposing intermediate value friction coefficients developed by a random number generator with Gaussian normal distribution characteristics. The assigned values are then held constant during the entire simulation in order to obtain reproducible results.† Thus, in this manner, the WPMR analysis results are brought closer to the realistic structural conditions.

The coefficient of friction (μ) between the pedestal supports and the pool floor is indeterminate. According to Rabinowicz [6.5.1], results of 199 tests performed on austenitic stainless steel plates submerged in water show a mean value of μ to be 0.503 with standard deviation of 0.125. Upper and lower bounds (based on twice standard deviation) are 0.753 and 0.253, respectively. Analyses are therefore performed for coefficient of friction values of 0.2 (lower limit), 0.8 (upper limit), and for random friction values clustered about a mean of 0.5. The bounding values of μ = 0.2 and 0.8 have been found to envelope the upper limit of module response in previous rerack projects.

6.6.4 Governing Equations of Motion

Using the structural model discussed in the foregoing, equations of motion corresponding to each degree-of-freedom are obtained using Lagrange's Formulation [6.6.1]. The system kinetic energy includes contributions from solid structures and from trapped and surrounding fluid. The final system of equations obtained have the matrix form:

$$[M]\left[\frac{d^2q}{dt^2}\right] = [Q] + [G]$$

† It is noted that DYNARACK has the capability to change the coefficient of friction at any pedestal at each instant of contact based on a random reading of the computer clock cycle. However, exercising this option would yield results that could not be reproduced. Therefore, the random choice of coefficients is made only once per run.

where:

[M] - total mass matrix (including structural and fluid mass contributions). The size of this matrix will be 22n x22n for a WPMR analysis (n = number of racks in the model).

q - the nodal displacement vector relative to the pool slab displacement (the term with q indicates the second derivative with respect to time, i.e., acceleration)

[G] - a vector dependent on the given ground acceleration

[Q] - a vector dependent on the spring forces (linear and nonlinear) and the coupling between degrees-of-freedom

The above column vectors have length 22n. The equations can be rewritten as follows:

$$\left[\frac{d^2 q}{d t^2} \right] = [M]^{-1} [Q] + [M]^{-1} [G]$$

This equation set is mass uncoupled, displacement coupled at each instant in time. The numerical solution uses a central difference scheme built into the proprietary computer program DYNARACK [6.2.4].

6.7 Structural Evaluation of Spent Fuel Rack

6.7.1 Kinematic and Stress Acceptance

There are two sets of criteria to be satisfied by the rack modules:

a. <u>Kinematic Criteria</u>

Per Reference [6.1.1], in order to be qualified as a physically stable structure it is necessary to demonstrate that an isolated rack in water would not overturn when an event of magnitude:

- 1.5 times the upset seismic loading condition is applied.
- 1.1 times the faulted seismic loading condition is applied.

b. Stress Limit Criteria

Stress limits must not be exceeded under the postulated load combinations provided herein.

6.7.2 Stress Limit Evaluations

The stress limits presented below apply to the rack structure and are derived from the ASME Code, Section III, Subsection NF [6.7.1]. Parameters and terminology are in accordance with the ASME Code. Material properties are obtained from the ASME Code Appendices [6.7.2], and are listed in Table 6.3.1.

(i) Normal and Upset Conditions (Level A or Level B)

a. Allowable stress in tension on a net section is:

$F_t = 0.6\ S_y$

Where, S_y = yield stress at temperature, and F_t is equivalent to primary membrane stress.

b. Allowable stress in shear on a net section is:

$F_v\ = .4\ S_y$

c. Allowable stress in compression on a net section

$$F_a = S_y\left(.47 - \frac{k\lambda}{444r}\right)$$

kλ/r for the main rack body is based on the full height and cross section of the honeycomb region and does not exceed 120 for all sections.

λ = unsupported length of component

k = length coefficient which gives influence of boundary conditions. The following values are appropriate for the described end conditions:

= 1 (simple support both ends)

= 2 (cantilever beam)

= 1/2 (clamped at both ends)
r = radius of gyration of component

d. Maximum allowable bending stress at the outermost fiber of a net section, due to flexure about one plane of symmetry is:

$F_b = 0.60\ S_y$ (equivalent to primary bending)

e. Combined bending and compression on a net section satisfies:

$$\frac{f_a}{F_a} + \frac{C_{mx}\ f_{bx}}{D_x\ F_{bx}} + \frac{C_{my}\ f_{by}}{D_y\ F_{by}} < 1$$

where:

f_a = Direct compressive stress in the section
f_{bx} = Maximum bending stress along x-axis
f_{by} = Maximum bending stress along y-axis
C_{mx} = 0.85
C_{my} = 0.85
D_x = $1 - (f_a/F'_{ex})$
D_y = $1 - (f_a/F'_{ey})$
$F'_{ex,ey}$ = $(\pi^2\ E)/(2.15\ (kl/r)^2_{x,y})$
E = Young's Modulus
and subscripts x,y reflect the particular bending plane.

f. Combined flexure and compression (or tension) on a net section:

$$\frac{f_a}{0.6\,S_y} + \frac{f_{bx}}{F_{bx}} + \frac{f_{by}}{F_{by}} < 1.0$$

The above requirements are to be met for both direct tension or compression.

g. Welds

Allowable maximum shear stress on the net section of a weld is given by:

$$F_w = 0.3\ S_u$$

where S_u is the weld material ultimate strength at temperature. For fillet weld legs in contact with base metal, the shear stress on the gross section is limited to $0.4S_y$.

where S_y is the base material yield strength at temperature.

(ii) Level D Service Limits

Section F-1334 (ASME Section III, Appendix F) [6.7.2], states that the limits for the Level D condition are the minimum of 1.2 (S_y/F_t) or $(0.7S_u/F_t)$ times the corresponding limits for the Level A condition. S_u is ultimate tensile stress at the specified rack design temperature. Examination of material properties for 304L stainless demonstrates that 1.2 times the yield strength is less than the 0.7 times the ultimate strength.

Exceptions to the above general multiplier are the following:

a) Stresses in shear shall not exceed the lesser of $0.72S_y$ or $0.42S_u$. In the case of the Austenitic Stainless material used here, $0.72S_y$ governs.

b) Axial Compression Loads shall be limited to 2/3 of the calculated buckling load.

c) Combined Axial Compression and Bending - The equations for Level A conditions shall apply except that:

   F_a = 0.667 x Buckling Load/ Gross Section Area,

   and the terms F'_{ex} and F'_{ey} may be increased by the factor 1.65.

d) For welds, the Level D allowable maximum weld stress is not specified in Appendix F of the ASME Code. An appropriate limit for weld throat stress is conservatively set here as:

   $F_w = (0.3\ S_u)$ x factor

   where:

   factor = (Level D shear stress limit)/(Level A shear stress limit)

6.7.3 Dimensionless Stress Factors

For convenience, the stress results are presented in dimensionless form. Dimensionless stress factors are defined as the ratio of the actual developed stress to the specified limiting value. The limiting value of each stress factor is 1.0, based on the allowable strengths for each level, for Levels A, B, and D (where $1.2S_y < .7S_u$). Stress factors reported are:

R_1 = Ratio of direct tensile or compressive stress on a net section to its allowable value (note pedestals only resist compression)

R_2 = Ratio of gross shear on a net section in the x-direction to its allowable value

R_3 = Ratio of maximum x-axis bending stress to its allowable value for the section

R_4 = Ratio of maximum y-axis bending stress to its allowable value for the section

R_5 = Combined flexure and compressive factor (as defined in the foregoing)

R_6 = Combined flexure and tension (or compression) factor (as defined in the foregoing)

R_7 = Ratio of gross shear on a net section in the y-direction to its allowable value

6.7.4 Loads and Loading Combinations for Spent Fuel Racks

The applicable loads and their combinations which must be considered in the seismic analysis of rack modules is excerpted from Refs. [6.1.2] and [6.6.3]. The load combinations considered are identified below:

Loading Combination	Service Level
$D + L$ $D + L + T_o$ $D + L + T_o + E$	Level A
$D + L + T_a + E$ $D + L + T_o + P_f$	Level B
$D + L + T_a + E'$ $D + L + T_o + F_d$	Level D The functional capability of the fuel racks must be demonstrated.

Where:

D	=	Dead weight-induced loads (including fuel assembly weight)
L	=	Live Load (not applicable for the fuel rack, since there are no moving objects in the rack load path)
P_f	=	Upward force on the racks caused by postulated stuck fuel assembly
F_d	=	Impact force from accidental drop of the heaviest load from the maximum possible height.
E	=	Operating Basis Earthquake (OBE)
E'	=	Safe Shutdown Earthquake (SSE)
T_o	=	Differential temperature induced loads (normal operating or shutdown condition based on the most critical transient or steady state condition)
T_a	=	Differential temperature induced loads (the highest temperature associated with the postulated abnormal design conditions)

T_a and T_o produce local thermal stresses. The worst thermal stress field in a fuel rack is obtained when an isolated storage location has a fuel assembly generating heat at maximum postulated rate and surrounding storage locations contain no fuel. Heated water makes unobstructed contact with the inside of the storage walls, thereby producing maximum possible temperature difference between adjacent cells. Secondary stresses produced are limited to the body of the rack; that is, support pedestals do not experience secondary (thermal) stresses.

6.8 Seismic Analysis

6.8.1 Acceptance Criteria

Only the SSE event based cases are selected for dynamic simulations if the maximum stress factors obtained from these cases are below the limit prescribed for OBE events. The maximum stress factor limit for OBE events is one half of the stress factor limit for SSE events. Therefore, if the stress factors obtained from the SSE cases are less than 0.5 then they also meet the OBE stress factor limits and hence no further OBE runs are required.

6.8.2 Parametric Simulations

Consideration of the parameters described earlier results in a number of scenarios for both the WPMR and the single rack analyses. Using the criterion presented in 6.8.1, the number of scenarios can be conservatively reduced. This analysis considers only SSE simulations since the results from these simulations meet the above acceptance criteria. Although not essential, one additional simulation (Run No. 7) is performed for comparison between the SSE and OBE results. This additional run is a re-run of most bounding SSE simulation with the OBE seismic time histories.

The table below presents a complete listing of the simulations discussed herein. The Whole Pool Multi-Rack model considers all fifteen new racks in the pool. In addition to this basic model, an interim configuration is also considered for the scenario when only the nine racks closest to the pool's West wall (see figure 2.1) are installed. This interim configuration is selected because of the large fluid gap, due to the absence of remaining new racks in the pool, weakens the hydrodynamic effect and, therefore, yields large rack displacements and pedestal loads. Rack number 3,4,5,8,9 and 10 (see Figure 6.8.1) are not considered in this model. The rack numbering scheme used to identify the racks for whole pool multi rack (WPMR) simulation is introduced in Figures 6.8.1 or 6.8.2. Single rack analyses are performed to investigate the structural adequacy of the rack when subjected to an array of different fuel loading patterns (for example Fully loaded, partially loaded, etc.) and interface coefficient of frictions. Single rack simulations are also used to confirm the WPMR results and to determine the potential for rack overturning. In the evaluations, one rack from each region was chosen for the single rack analysis. Rack C1 (Region 2) and Rack D5 (Region 1) were selected, as they are the most slender, i.e. they have the highest aspect ratios in their respective regions. In addition to these single rack simulations, two single rack runs that exhibit the greatest displacement are re-run with severe earthquake conditions (1.5xSSE) for the purpose of checking the potential for rack overturning. Run no. 20 and 33 are such runs.

LIST OF RACK SIMULATIONS				
Run	**Model**	**Load Case**	**COF**	**Event**
1	WPMR	Full Pool	0.2	SSE

LIST OF RACK SIMULATIONS				
Run	**Model**	**Load Case**	**COF**	**Event**
2	WPMR	Full Pool	0.8	SSE
3	WPMR	Full Pool	Random	SSE
4	WPMR	Interim Configuration	0.2	SSE
5	WPMR	Interim Configuration	0.8	SSE
6	WPMR	Interim Configuration	Random	SSE
7	WPMR	Full Pool	Random	OBE
SINGLE RACK RUNS (Rack C1)				
8	Single Rack	Fully Loaded	0.2	SSE
9	Single Rack	Fully Loaded	0.8	SSE
10	Single Rack	Fully Loaded	0.5	SSE
11	Single Rack	Nearly Empty	0.2	SSE
12	Single Rack	Nearly Empty	0.8	SSE
13	Single Rack	Nearly Empty	0.5	SSE
14	Single Rack	Half-Full Rack (symmetric about diagonal)	0.2	SSE
15	Single Rack	Half-Full Rack (symmetric about diagonal)	0.8	SSE
16	Single Rack	Half-Full Rack (symmetric about diagonal)	0.5	SSE
17	Single Rack	Half-Full Rack (symmetric about short axis)	0.2	SSE
18	Single Rack	Half-Full Rack (symmetric about short axis)	0.8	SSE
19	Single Rack	Half-Full Rack (symmetric about short axis)	0.5	SSE

LIST OF RACK SIMULATIONS				
Run	**Model**	**Load Case**	**COF**	**Event**
20	Single Rack	Case with max. Displacement	0.5	1.5 x SSE

SINGLE RACK RUNS (Rack D5)				
21	Single Rack	Fully Loaded	0.2	SSE
22	Single Rack	Fully Loaded	0.8	SSE
23	Single Rack	Fully Loaded	0.5	SSE
24	Single Rack	Nearly Empty	0.2	SSE
25	Single Rack	Nearly Empty	0.8	SSE
26	Single Rack	Nearly Empty	0.5	SSE
27	Single Rack	Half-Full Rack (symmetric about diagonal)	0.2	SSE
28	Single Rack	Half-Full Rack (symmetric about diagonal)	0.8	SSE
29	Single Rack	Half-Full Rack (symmetric about diagonal)	0.5	SSE
30	Single Rack	Half-Full Rack (symmetric about short axis)	0.2	SSE
31	Single Rack	Half-Full Rack (symmetric about short axis)	0.8	SSE
32	Single Rack	Half-Full Rack (symmetric about short axis)	0.5	SSE
33	Single Rack	Case with max. displacement	0.5	1.5 x SSE

Where:

Random = Gaussian distribution with a mean of 0.5 coefficient of friction (upper and lower limits of 0.2 and 0.8)

Note that run no. 20 and 33 are re-runs of run no. 10 and 23 except that the racks in these runs are simulated as an isolated rack in the pool as required by subsection 6.7.1.

6.9 Time History Simulation Results

The results from the DYNARACK runs may be seen in the raw data output files. However, due to the huge quantity of output data, a post-processor is used to scan for worst case conditions and develop the stress factors. Further reduction in this bulk of information is provided in this section by extracting the worst case values from the parameters of interest; namely displacements, support pedestal forces, impact loads, and stress factors. This section also summarizes other analyses performed to develop and evaluate structural member stresses, which are not determined by the post processor. For each table, the COF column refers to the interface coefficient of friction discussed in subsection 6.2.1. The "Rack" column denotes racks by number (applicable to the DYNARACK model) for whole pool multi rack runs and by letter (applicable to the pool layout drawing) for single rack runs.

6.9.1 Rack Displacements

A tabulated summary of the maximum displacement for each simulation is provided below. Note that all of the maximum displacements occurred at the tops of the storage racks, as expected from swaying, bending, and tipping behavior. The location/direction terms defined as follows:

uxt, uyt = displacement of top corner of rack, relative to the slab, in the North-South and East-West directions, respectively. The maximum displacements for every simulation, including the single rack tipover analyses, occurred at the top of the racks shown in the last table column.

RACK DISPLACEMENT RESULTS

Run	Model	COF	Max. Displacement (inches)	Location/ (x or y) Direction	Rack
1	WPMR (full)	0.2	0.747	Top	13
2	WPMR (full)	0.8	0.75	Top	9
3	WPMR (full)	Random	0.743	Top	5
4	WPMR (interim)	0.2	0.645	Top	2

RACK DISPLACEMENT RESULTS					
Run	**Model**	**COF**	**Max. Displacement (inches)**	**Location/ (x or y) Direction**	**Rack**
5	WPMR (interim)	0.8	1.03	Top	6
6	WPMR (interim)	Random	0.745	Top	6
7	WPMR (full)	Random	0.422	Top	13
SINGLE RACK RUNS (Rack C1)					
8	single rack (full)	0.2	0.38	Top	C1
9	single rack (full)	0.8	0.4193	Top	C1
10	single rack (full)	0.5	0.4254	Top	C1
11	single rack (nearly empty)	0.2	0.0719	Top	C1
12	single rack (nearly empty)	0.8	0.0714	Top	C1
13	single rack (nearly empty)	0.5	0.073	Top	C1
14	single rack (half)	0.2	0.235	Top	C1
15	single rack (half)	0.8	0.2851	Top	C1
16	single rack (half)	0.5	0.283	Top	C1
17	single rack (half-short axis)	0.2	0.2153	Top	C1
18	single rack (half-short axis)	0.8	0.2371	Top	C1
19	single rack (half-short axis)	0.5	0.2387	Top	C1
20	single rack (overturning)	0.5	0.492	Top	C1
SINGLE RACK RUNS (Rack D5)					
21	single rack (full)	0.2	0.266	Top	D5
22	single rack (full)	0.8	0.382	Top	D5
23	single rack (full)	0.5	0.4062	Top	D5
24	single rack (nearly empty)	0.2	0.0848	Top	D5
25	single rack (nearly empty)	0.8	0.107	Top	D5

RACK DISPLACEMENT RESULTS					
Run	**Model**	**COF**	**Max. Displacement (inches)**	**Location/ (x or y) Direction**	**Rack**
26	single rack (nearly empty)	0.5	0.1098	Top	D5
27	single rack (half)	0.2	0.283	Top	D5
28	single rack (half)	0.8	0.546	Top	D5
29	single rack (half)	0.5	0.514	Top	D5
30	single rack (half-short axis)	0.2	0.1594	Top	D5
31	single rack (half-short axis)	0.8	0.217	Top	D5
32	single rack (half-short axis)	0.5	0.2086	Top	D5
33	single rack (overturning)	0.5	1.02	Top	D5

The table shows that the maximum rack displacement is only 1.03 inches which occurs during run No. 5. This small displacement indicates that rack overturning is not a concern.

6.9.2 Pedestal Vertical Forces

Pedestal number 1 for each rack is located in the northeast corner of the rack. Numbering increases counterclockwise around the periphery of each rack. The following bounding vertical pedestal forces are obtained for each run:

MAXIMUM VERTICAL LOADS					
Run	**Model**	**COF**	**Event**	**Max. Vertical Load**	**Rack**
1	WPMR (full)	0.2	SSE	127000	10
2	WPMR (full)	0.8	SSE	146000	12
3	WPMR (full)	Random	SSE	147000	6
4	WPMR (interim)	0.2	SSE	125000	1
5	WPMR (interim)	0.8	SSE	143000	7
6	WPMR (interim)	Random	SSE	145000	1
7	WPMR (full)	Random	OBE	124000	11
SINGLE RACK RUNS (Rack C1)					
8	single rack (full)	0.2	SSE	101000	C1
9	single rack (full)	0.8	SSE	108000	C1
10	single rack (full)	0.5	SSE	109000	C1
11	single rack (nearly empty)	0.2	SSE	16400	C1
12	single rack (nearly empty)	0.8	SSE	18100	C1
13	single rack (nearly empty)	0.5	SSE	18100	C1
14	single rack (half)	0.2	SSE	60600	C1
15	single rack (half)	0.8	SSE	61400	C1
16	single rack (half)	0.5	SSE	61500	C1
17	single rack (half-short axis)	0.2	SSE	54400	C1
18	single rack (half-short axis)	0.8	SSE	67000	C1
19	single rack (half-short axis)	0.5	SSE	67000	C1
20	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A
SINGLE RACK RUNS (Rack D5)					

MAXIMUM VERTICAL LOADS					
Run	**Model**	**COF**	**Event**	**Max. Vertical Load**	**Rack**
21	single rack (full)	0.2	SSE	104000	D5
22	single rack (full)	0.8	SSE	113000	D5
23	single rack (full)	0.5	SSE	111000	D5
24	single rack (nearly empty)	0.2	SSE	20900	D5
25	single rack (nearly empty)	0.8	SSE	27900	D5
26	single rack (nearly empty)	0.5	SSE	27800	D5
27	single rack (half)	0.2	SSE	67700	D5
28	single rack (half)	0.8	SSE	84400	D5
29	single rack (half)	0.5	SSE	78800	D5
30	single rack (half-short axis)	0.2	SSE	60800	D5
31	single rack (half-short axis)	0.8	SSE	70000	D5
32	single rack (half-short axis)	0.5	SSE	69800	D5
33	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A

As may be seen, the highest pedestal load of 147,000 lbs. occurs in run 3 of the WPMR model. The effect of this load is evaluated in the bearing pad analysis.

6.9.3 Pedestal Friction Forces

The maximum (x or y direction) shear load bounding all pedestals in the simulation are reported below and are obtained by inspection of the complete tabular data.

MAXIMUM HORIZONTAL LOADS					
Run	**Model**	**COF**	**Event**	**Max. Horizontal Load**	**Rack**
1	WPMR (full)	0.2	SSE	23800	11
2	WPMR (full)	0.8	SSE	51700	6
3	WPMR (full)	Random	SSE	46900	6
4	WPMR (interim)	0.2	SSE	23500	1
5	WPMR (interim)	0.8	SSE	43200	11
6	WPMR (interim)	Random	SSE	39800	11
7	WPMR (full)	Random	OBE	33300	13
SINGLE RACK RUNS (Rack C1)					
8	single rack (full)	0.2	SSE	17900	C1
9	single rack (full)	0.8	SSE	35600	C1
10	single rack (full)	0.5	SSE	30600	C1
11	single rack (nearly empty)	0.2	SSE	2800	C1
12	single rack (nearly empty)	0.8	SSE	5380	C1
13	single rack (nearly empty)	0.5	SSE	5420	C1
14	single rack (half)	0.2	SSE	11300	C1
15	single rack (half)	0.8	SSE	18500	C1
16	single rack (half)	0.5	SSE	18800	C1
17	single rack (half-short axis)	0.2	SSE	8890	C1
18	single rack (half-short axis)	0.8	SSE	24800	C1
19	single rack (half-short axis)	0.5	SSE	22800	C1
20	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A
SINGLE RACK RUNS (Rack D5)					

MAXIMUM HORIZONTAL LOADS					
Run	**Model**	**COF**	**Event**	**Max. Horizontal Load**	**Rack**
21	single rack (full)	0.2	SSE	17600	D5
22	single rack (full)	0.8	SSE	31000	D5
23	single rack (full)	0.5	SSE	31200	D5
24	single rack (nearly empty)	0.2	SSE	3960	D5
25	single rack (nearly empty)	0.8	SSE	8640	D5
26	single rack (nearly empty)	0.5	SSE	8530	D5
27	single rack (half)	0.2	SSE	12700	D5
28	single rack (half)	0.8	SSE	19700	D5
29	single rack (half)	0.5	SSE	19000	D5
30	single rack (half-short axis)	0.2	SSE	10600	D5
31	single rack (half-short axis)	0.8	SSE	24900	D5
32	single rack (half-short axis)	0.5	SSE	19400	D5
33	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A

The largest pedestal load of 51,700 lbs occurs in run 2 of the WPMR model. The effect of this load is evaluated in the liner fatigue analysis.

6.9.4 Rack Impact Loads

A freestanding rack, by definition, is a structure subject to potential impacts during a seismic event. Impacts arise from rattling of the fuel assemblies in the storage rack locations and, in some instances, from localized impacts between the racks, or between a peripheral rack and the pool wall. The following sections discuss the bounding values of these impact loads.

6.9.4.1 Rack to Rack Impacts

There is no rack to rack impact at rack top between any two racks during any of the seismic events. However, rack to rack impacts at the baseplate of the structure are predicted between Holtec racks. There are no impacts between Holtec racks and Westinghouse racks during any of the seismic events. The maximum instantaneous impact forces at the baseplate are summarized below from all simulations performed.

MAXIMUM RACK-TO-RACK (BASEPLATE) IMPACT		
Run	**Model**	**Max. Impact Load (kips)**
4	WPMR	20.67
21	Single	6.28

It may be noted that all impact loads occurred only at the bottom of the racks.

6.9.4.2 Rack to Wall Impacts

Racks did not impact the pool walls under any simulation.

6.9.4.3 Fuel to Cell Wall Impact Loads

A review of all simulations performed allows determination of the maximum instantaneous impact load between fuel assembly and fuel cell wall at any modeled impact site. The maximum fuel/cell wall impact load values are reported in the following table.

FUEL-TO-CELL WALL IMPACT					
Run	**Model**	**COF**	**Event**	**Max. Impact Load (lbs)**	**Rack**
1	WPMR (full)	0.2	SSE	802	15
2	WPMR (full)	0.8	SSE	752	15
3	WPMR (full)	Random	SSE	697	15
4	WPMR (interim)	0.2	SSE	630	15
5	WPMR (interim)	0.8	SSE	592	2
6	WPMR (interim)	Random	SSE	592	2
7	WPMR (full)	Random	OBE	432	9
SINGLE RACK RUNS (Rack C1)					
8	single rack (full)	0.2	SSE	460	C1
9	single rack (full)	0.8	SSE	443	C1
10	single rack (full)	0.5	SSE	426	C1
11	single rack (nearly empty)	0.2	SSE	510	C1
12	single rack (nearly empty)	0.8	SSE	510	C1
13	single rack (nearly empty)	0.5	SSE	510	C1
14	single rack (half)	0.2	SSE	530	C1
15	single rack (half)	0.8	SSE	533	C1
16	single rack (half)	0.5	SSE	533	C1
17	single rack (half-short axis)	0.2	SSE	517	C1
18	single rack (half-short axis)	0.8	SSE	523	C1
19	single rack (half-short axis)	0.5	SSE	523	C1
20	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A
SINGLE RACK RUNS (Rack D5)					
21	single rack (full)	0.2	SSE	453	D5

FUEL-TO-CELL WALL IMPACT					
Run	**Model**	**COF**	**Event**	**Max. Impact Load (lbs)**	**Rack**
22	single rack (full)	0.8	SSE	462	D5
23	single rack (full)	0.5	SSE	462	D5
24	single rack (nearly empty)	0.2	SSE	488	D5
25	single rack (nearly empty)	0.8	SSE	450	D5
26	single rack (nearly empty)	0.5	SSE	450	D5
27	single rack (half)	0.2	SSE	473	D5
28	single rack (half)	0.8	SSE	478	D5
29	single rack (half)	0.5	SSE	480	D5
30	single rack (half-short axis)	0.2	SSE	520	D5
31	single rack (half-short axis)	0.8	SSE	477	D5
32	single rack (half-short axis)	0.5	SSE	477	D5
33	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A

The maximum Fuel-to-Cell Wall Impact is recorded to be 802 lbs. during run no. 1. The structural integrity of the cell wall under the impact of this load must be evaluated. The discussion of this evaluation is provided in Section 6.10.3.

[illegible]	single rack (half-short axis)	0.5	[illegible]	Top	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	Top	[illegible]
[illegible] RUNS [illegible]					
[illegible]	single rack (full)	0.2	[illegible]	Top	[illegible]
22	single rack (full)	0.8	[illegible]	Top	[illegible]
23	[illegible]	0.5	[illegible]	Top	[illegible]
[illegible]	[illegible]	0.2	[illegible]	[illegible]	[illegible]

[illegible]					
[illegible]	[illegible]	COF	[illegible]	[illegible]	Rack
[illegible]	[illegible]	0.5	[illegible]	Top	D5
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	D5
[illegible]	[illegible]	0.8	[illegible]	Top	D5
29	[illegible]	[illegible]	[illegible]	Top	D5

6.10 Rack Structural Evaluation

6.10.1 Rack Stress Factors

With time history results available for pedestal normal and lateral interface forces, the maximum values for the previously defined stress factors can be determined for every pedestal in the array of racks. With this information available, the structural integrity of the pedestal can be assessed and reported. The net section maximum (in time) bending moments and shear forces can also be determined at the bottom casting-rack cellular structure interface for each spent fuel rack in the pool. With this information in hand, the maximum stress in the limiting rack cell (box) can be evaluated.

From all of the simulations, the bounding stress factors for each run, in either cellular or the pedestal region, are summarized below :

It is evident from the DYNARACK results for the stress factors that the maximum stresses occur at the cellular region to the baseplate (CRB) interface. The compressive stress in the CRB is principally due to the flexural motion of the rack module. In order to account for the possible reduction in the section modulus of the CRB section due to the localized compressive stress, we assume that the maximum compressive stress occurs over the entire CRB section. With this extremely conservative assumption, the stress magnifier per NF-3222.2 can be calculated and applied to the stress factors of the DYNARACK results. The table below incorporates the slenderness magnifier on the CRB section compressive stress values called from the DYNARACK runs.

MAXIMUM STRESS FACTORS

Run	Model	COF	Event	Stress Factor Cell (CRB)	Stress Factor Type/Rack
1	WPMR (full)	0.2	SSE	0.367	R6/5
2	WPMR (full)	0.8	SSE	0.403	R6/5
3	WPMR (full)	Random	SSE	0.401	R6/5
4	WPMR (interim)	0.2	SSE	0.377	R6/2
5	WPMR (interim)	0.8	SSE	0.378	R6/7
6	WPMR (interim)	Random	SSE	0.383	R6/7
7	WPMR (full)	Random	OBE	0.522	R6/10
SINGLE RACK RUNS (Rack C1)					
8	single rack (full)	0.2	SSE	0.312	R6/C1
9	single rack (full)	0.8	SSE	0.340	R6/C1
10	single rack (full)	0.5	SSE	0.343	R6/C1
11	single rack (nearly empty)	0.2	SSE	0.051	R6/C1
12	single rack (nearly empty)	0.8	SSE	0.055	R6/C1
13	single rack (nearly empty)	0.5	SSE	0.055	R6/C1
14	single rack (half)	0.2	SSE	0.189	R6/C1
15	single rack (half)	0.8	SSE	0.191	R6/C1
16	single rack (half)	0.5	SSE	0.191	R6/C1
17	single rack (half-short axis)	0.2	SSE	0.168	R6/C1
18	single rack (half-short axis)	0.8	SSE	0.202	R6/C1
19	single rack (half-short axis)	0.5	SSE	0.202	R6/C1
20	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A

SINGLE RACK RUNS (Rack D5)					
21	single rack (full)	0.2	SSE	0.196	R6/D5
22	single rack (full)	0.8	SSE	0.208	R6/D5
23	single rack (full)	0.5	SSE	0.209	R6/D5
24	single rack (nearly empty)	0.2	SSE	0.040	R6/D5
25	single rack (nearly empty)	0.8	SSE	0.050	R6/D5
26	single rack (nearly empty)	0.5	SSE	0.050	R6/D5
27	single rack (half)	0.2	SSE	0.129	R6/D5
28	single rack (half)	0.8	SSE	0.154	R6/D5
29	single rack (half)	0.5	SSE	0.144	R6/D5
30	single rack (half-short axis)	0.2	SSE	0.117	R6/D5
31	single rack (half-short axis)	0.8	SSE	0.129	R6/D5
32	single rack (half-short axis)	0.5	SSE	0.131	R6/D5
33	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A

Thus, the maximum stress factor in either pedestal or cellular region for SSE and OBE are 0.403 and 0.522, respectively. An evaluation of the stress factors for all of the simulations performed, leads to the conclusion that all stress factors are less than the mandated limit of 1.0 for the load cases examined. The stress allowables are indeed satisfied for the load levels considered for every limiting location in every rack in the array.

6.10.2 Pedestal Thread Shear Stress

The complete post-processor results give thread stresses under faulted conditions for every pedestal for every rack in the pool. The average shear stress in the engagement region is given below for the limiting pedestal in each simulation.

THREAD SHEAR STRESS					
Run	**Model**	**COF**	**Event**	**Stress (psi)**	**Rack**
1	WPMR (full)	0.2	SSE	6162	10
2	WPMR (full)	0.8	SSE	7083	12
3	WPMR (full)	Random	SSE	7132	6
4	WPMR (interim)	0.2	SSE	6066	1
5	WPMR (interim)	0.8	SSE	6938	7
6	WPMR (interim)	Random	SSE	7035	1
7	WPMR (full)	Random	OBE	6016	11
SINGLE RACK RUNS (Rack C1)					
8	single rack (full)	0.2	SSE	4900	C1
9	single rack (full)	0.8	SSE	5240	C1
10	single rack (full)	0.5	SSE	5289	C1
11	single rack (nearly empty)	0.2	SSE	796	C1
12	single rack (nearly empty)	0.8	SSE	878	C1
13	single rack (nearly empty)	0.5	SSE	878	C1
14	single rack (half)	0.2	SSE	2940	C1
15	single rack (half)	0.8	SSE	2980	C1
16	single rack (half)	0.5	SSE	2984	C1
17	single rack (half-short axis)	0.2	SSE	2640	C1
18	single rack (half-short axis)	0.8	SSE	3251	C1
19	single rack (half-short axis)	0.5	SSE	3250	C1
20	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A
SINGLE RACK RUNS (Rack D5)					
21	single rack (full)	0.2	SSE	5046	D5

THREAD SHEAR STRESS					
Run	Model	COF	Event	Stress (psi)	Rack
22	single rack (full)	0.8	SSE	5483	D5
23	single rack (full)	0.5	SSE	5386	D5
24	single rack (nearly empty)	0.2	SSE	1014	D5
25	single rack (nearly empty)	0.8	SSE	1354	D5
26	single rack (nearly empty)	0.5	SSE	1349	D5
27	single rack (half)	0.2	SSE	3285	D5
28	single rack (half)	0.8	SSE	4095	D5
29	single rack (half)	0.5	SSE	3823	D5
30	single rack (half-short axis)	0.2	SSE	2950	D5
31	single rack (half-short axis)	0.8	SSE	3396	D5
32	single rack (half-short axis)	0.5	SSE	3387	D5
33	single rack (overturning)	0.5	1.5 x SSE	N/A	N/A

The ultimate strength of the female part of the pedestal is 66,200 psi. The yield stress for this material is 21,300 psi. The allowable shear stress for Level B (OBE) conditions is 0.4 times the yield stress which gives 8.520 psi and the allowable shear stress for level D is 0.72 times the yield stress which gives 15.336 psi. The maximum calculated shear stress value for the SSE is 7,132 psi and 6,016 psi for the OBE which are less than their respective allowable values. Therefore, thread shear stresses are acceptable under all conditions.

6.10.3 Local Stresses Due to Impacts

Impact loads at the pedestal base (discussed in subsection 6.9.2) produce stresses in the pedestal for which explicit stress limits are prescribed in the Code. However, impact loads on the cellular region of the racks, as discussed in subsection 6.9.4.3 above, produce stresses which attenuate rapidly away from the loaded region. This behavior is characteristic of secondary stresses.

Even though limits on secondary stresses are not prescribed in the Code for class 3 NF structures, evaluations must be made to ensure that the localized impacts do not lead to plastic deformations in the storage cells which affect the subcriticality of the stored fuel array.

a. <u>Impact Loading Between Fuel Assembly and Cell Wall</u>

Local cell wall integrity is conservatively estimated from peak impact loads. Plastic analysis is used to obtain the limiting impact load which would lead to gross permanent deformation. Table 6.9.1 indicates that the limiting impact load (of 3,187 lbf, including a safety factor of 2.0) is much greater than the highest calculated impact load value (of 802 lbf, see subsection 6.9.4.3) obtained from any of the rack analyses. Therefore, fuel impacts do not represent a significant concern with respect to fuel rack cell deformation.

b. <u>Impacts Between Adjacent Racks</u>

As may be seen from subsection 6.9.4.1, the bottom of the storage racks impact each other at a few locations during seismic events. Since the loading is presented edge-on to the 3/4" baseplate membrane, the distributed stresses after local deformation will be negligible. The impact loading will be distributed over a large area (a significant portion of the entire baseplate length of about 63 inches by its 3/4-inch thickness) . The resulting compressive stress from the highest impact load of 20,670 lbs. distributed over 47.25 sq. inches is only 438 psi, which is negligible. This is a conservative computation, since the simulation assumes a local impact site. Therefore, any deformation will not effect the configuration of the stored fuel. Impact between the racks in the cellular region containing active fuel is shown not to occur.

6.10.4 Assessment of Rack Fatigue Margin

Deeply submerged high density spent fuel storage racks arrayed in close proximity to each other in a free-standing configuration behave primarily as a nonlinear cantilevered structure when subjected to 3-D seismic excitations. In addition to the pulsations in the vertical load at each pedestal, lateral friction forces at the pedestal/bearing pad-liner interface, which help prevent or mitigate lateral sliding of the rack, also exert a time-varying moment in the baseplate region of the rack. The friction-induced lateral forces act simultaneously in x and y directions with the requirement that their vectorial sum does not exceed μN, where μ is the limiting interface coefficient of friction and N is the concomitant vertical thrust on the liner (at the *given* time instant). As the vertical thrust at a pedestal location changes, so does the maximum friction force, F, that the interface can exert. In other words, the lateral force at the pedestal/liner interface, F, is given by

$$F \leq \mu N(\tau)$$

where N (vertical thrust) is the time-varying function of τ. F does not always equal μN; rather, μN is the maximum value it can attain at any time; the actual value, of course, is determined by the dynamic equilibrium of the rack structure. In summary, the horizontal friction force at the pedestal/liner interface is a function of time; its magnitude and direction of action varies during the earthquake event.

The time-varying lateral (horizontal) and vertical forces on the extremities of the support pedestals produce stresses at the root of the pedestals in the manner of an end-loaded cantilever. The stress field in the cellular region of the rack is quite complex, with its maximum values located in the region closest to the pedestal. The maximum magnitude of the stresses depends on the severity of the pedestal end loads and on the geometry of the pedestal/rack baseplate region.

Alternating stresses in metals produce metal fatigue if the amplitude of the stress cycles is sufficiently large. In high density racks designed for sites with moderate to high postulated seismic action, the stress intensity amplitudes frequently reach values above the material endurance limit, leading to expenditure of the fatigue "usage" reserve in the material.

Because the locations of maximum stress (viz., the pedestal/rack baseplate junction) and the close placement of racks, a post-earthquake inspection of the high stressed regions in the racks is not feasible. Therefore, the racks must be engineered to withstand multiple earthquakes without reliance of nondestructive inspections for post-earthquake integrity assessment. The fatigue life evaluation of racks is an integral aspect of a sound design.

The time-history method of analysis, deployed in this report, provides the means to obtain a complete cycle history of the stress intensities in the highly stressed regions of the rack. Having determined the amplitude of the stress intensity cycles and their number, the cumulative damage factor, U, can be determined using the classical Miner's rule

$$U = \Sigma \frac{n_i}{N_i}$$

where n_i is the number of stress intensity cycles of amplitude σ_i, and N_i is the permissible number of cycles corresponding to σ_i from the ASME fatigue curve for the material of construction. U must be less than or equal to 1.0.

To evaluate the cumulative damage factor, a finite element model of a portion of the spent fuel rack in the vicinity of a support pedestal is constructed in sufficient detail to provide an accurate assessment of stress intensities. Figure 6.10.1 shows the essentials of the finite element model. The finite element solutions for unit pedestal loads in three orthogonal directions are combined to establish the maximum value of stress intensity as a function of the three unit pedestal loads. Using the archived results of the spent fuel rack dynamic analyses (pedestal load histories versus time), enables a time-history of stress intensity to be established at the most limiting location. This permits establishing a set of alternating stress intensity ranges versus cycles for an SSE and an OBE event. Following ASME Code guidelines for computing U, it is found that U =0.92 due to the combined effect of one SSE and twenty OBE events. This is below the ASME Code limit of 1.0.

6.10.5 Weld Stresses

Weld locations subjected to significant seismic loading are at the bottom of the rack at the baseplate-to-cell connection, at the top of the pedestal support at the baseplate connection, and at cell-to-cell connections. Bounding values of resultant loads are used to qualify the connections. Table 6.9.1 provides the comparison of calculated stress vs. allowable stress.

a. Baseplate-to-Rack Cell Welds

The highest predicted weld stress for SSE is calculated from the set of forces Fx, Fy and Fz at the Cell Baseplate interface when R6 (defined above in 6.10.1) is maximum. The weld between the cell and the baseplate is checked to determine that the maximum weld stress under SSE event is 11,520 psi. This value is less than the permissible allowable value of 35,748 psi.

b. Baseplate-to-Pedestal Welds

The weld between baseplate and support pedestal is checked to determine that the maximum stress under the SSE and the OBE event are 6,975 psi and 4,194 psi respectively. These calculated stress values are well below the SSE and OBE allowable of 35,748 psi and 19,860 psi, respectively.

c. Cell-to-Cell Welds

Cell-to-cell connections are formed by a series of connecting welds along the cell height. Stresses in storage cell to cell welds develop due to fuel assembly impacts with the cell wall. These weld stresses are conservatively calculated by assuming that fuel assemblies in adjacent cells are moving out of phase with one another so that impact loads in two adjacent cells are in opposite directions; this tends to separate the two cells from each other at the weld. Table 6.9.1 gives results for the maximum allowable load that can be transferred by these welds based on the available weld area. An upper bound on the load required to be transferred is also given in Table 6.9.1 and is much lower than the allowable load. This upper bound value is very conservatively obtained by applying the bounding rack-to-fuel impact load from any simulation in two orthogonal directions simultaneously, and multiplying the result by 2 to account for the simultaneous impact of two assemblies. An equilibrium analysis at the connection then yields the upper bound load to be transferred. It is seen from the results in Table 6.9.1 that the calculated load is well below the allowable load.

6.11 Level A Evaluation

The Level A condition is not a governing condition for spent fuel racks since the general level of loading is far less than Level B loading. To illustrate this, the heaviest spent fuel rack is considered under the dead weight load. It is shown below that the maximum pedestal load is low and that further stress evaluations are unnecessary.

LEVEL A MAXIMUM PEDESTAL LOAD

Dry Weight of Largest Holtec Rack (Region 1)	=	18,050 lbf
Dry Weight of 70 Fuel Assemblies	=	119,000 lbf
Total Dry Weight	=	137,050 lbf
Total Buoyant Weight (0.87 × Total Dry Weight)	=	119,233.5 lbf
Load per Pedestal	=	29,808 lbf

The stress allowables for the normal condition is the same as for the upset condition, which resulted in a maximum pedestal load of 147,000 lbs. Since this load (and the corresponding stress throughout the rack members) is much greater than the 29,808 lb load calculated above, the seismic condition controls over normal (Gravity) condition. Therefore, no further evaluation is performed.

6.12 Hydrodynamic Loads on Pool Walls

The maximum hydrodynamic pressures (in psi) that develop between the fuel racks and the spent fuel pool walls develops for the case of the rack that exhibits the largest displacement. This has been done for both the SSE and OBE cases. The results for these worst case conditions are shown in the table below.

Case	Maximum Pressure (psi)
SSE	7.92
OBE	4.31

These hydrodynamic pressures were considered in the evaluation of the Spent Fuel Pool structure.

6.13 Conclusion

Thirty-three discrete freestanding dynamic simulations of maximum density spent fuel storage racks have been performed to establish the structural margins of safety. Of the thirty –three parametric analyses, four simulations consisted of modeling all 15 fuel racks in the pool in one comprehensive Whole Pool Multi Rack (WPMR) model. Three additional runs were performed for interim configuration case. The remaining twenty-six runs were carried out with the classical single rack 3-D model. The parameters varied in the different runs consisted of the rack/pool liner interface coefficient of friction, extent of storage locations occupied by spent nuclear fuel (ranging from nearly empty to full) and the type of seismic input (SSE or OBE). Maximum (maximum in time and space) values of pedestal vertical, shear forces, displacements and stress factors (normalized stresses for NF class 3 linear type structures) have been post-processed from the array of runs and summarized in tables in this chapter. The results show that:

(i) All stresses are well below their corresponding "NF" limits.

(ii) There is no rack-to-rack or rack-to-wall impact anywhere in the cellular region of the rack modules

(iii) The rack overturning is not a concern.

An evaluation of the fatigue expenditure in the most stressed location in the most heavily loaded rack module under combined effect of one SSE and twenty OBE events shows that the Cumulative Damage Factor (using Miner's rule) is below the permissible value of 1.0.

In conclusion, all evaluations of structural safety, mandated by the OT Position Paper [6.1.2] and the contemporary fuel rack structural analysis practice have been carried out. They demonstrate consistently large margins of safety in all new storage modules.

6.14 References

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[6.5.2] Singh, K.P. and Soler, A.I., "Dynamic Coupling in a Closely Spaced Two-Body System Vibrating in Liquid Medium: The Case of Fuel Racks," 3rd International Conference on Nuclear Power Safety, Keswick, England, May 1982.

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[6.7.2] ASME Boiler & Pressure Vessel Code, Section III, Appendices, 1995 Edition.

[6.7.3] USNRC Standard Review Plan, NUREG-0800 (Section 3.8.4, Rev. 2, 1989).

[6.10.1] Chun, R., Witte, M. and Schwartz, M., "Dynamic Impact Effects on Spent Fuel Assemblies," UCID-21246, Lawrence Livermore National Laboratory, October 1987.

Table 6.2.1

PARTIAL LISTING OF FUEL RACK APPLICATIONS USING DYNARACK

PLANT	DOCKET NUMBER(s)	YEAR
Enrico Fermi Unit 2	USNRC 50-341	1980
Quad Cities 1 & 2	USNRC 50-254, 50-265	1981
Rancho Seco	USNRC 50-312	1982
Grand Gulf Unit 1	USNRC 50-416	1984
Oyster Creek	USNRC 50-219	1984
Pilgrim	USNRC 50-293	1985
V.C. Summer	USNRC 50-395	1984
Diablo Canyon Units 1 & 2	USNRC 50-275, 50-323	1986
Byron Units 1 & 2	USNRC 50-454, 50-455	1987
Braidwood Units 1 & 2	USNRC 50-456, 50-457	1987
Vogtle Unit 2	USNRC 50-425	1988
St. Lucie Unit 1	USNRC 50-335	1987
Millstone Point Unit 1	USNRC 50-245	1989
Chinshan	Taiwan Power	1988
D.C. Cook Units 1 & 2	USNRC 50-315, 50-316	1992
Indian Point Unit 2	USNRC 50-247	1990
Three Mile Island Unit 1	USNRC 50-289	1991
James A. FitzPatrick	USNRC 50-333	1990
Shearon Harris Unit 2	USNRC 50-401	1991
Hope Creek	USNRC 50-354	1990
Kuosheng Units 1 & 2	Taiwan Power Company	1990
Ulchin Unit 2	Korea Electric Power Co.	1990
Laguna Verde Units 1 & 2	Comision Federal de Electricidad	1991
Zion Station Units 1 & 2	USNRC 50-295, 50-304	1992

Table 6.2.1

PARTIAL LISTING OF FUEL RACK APPLICATIONS USING DYNARACK

Sequoyah	USNRC 50-327, 50-328	1992
LaSalle Unit 1	USNRC 50-373	1992
Duane Arnold Energy Center	USNRC 50-331	1992
Fort Calhoun	USNRC 50-285	1992
Nine Mile Point Unit 1	USNRC 50-220	1993
Beaver Valley Unit 1	USNRC 50-334	1992
Salem Units 1 & 2	USNRC 50-272, 50-311	1993
Limerick	USNRC 50-352, 50-353	1994
Ulchin Unit 1	KINS	1995
Yonggwang Units 1 & 2	KINS	1996
Kori-4	KINS	1996
Connecticut Yankee	USNRC 50-213	1996
Angra Unit 1	Brazil	1996
Sizewell B	United Kingdom	1996

Table 6.3.1

RACK MATERIAL DATA (200°F)

(ASME - Section II, Part D)

Material	Young's Modulus E (psi)	Yield Strength S_y (psi)	Ultimate Strength S_u (psi)
SA240; 304L S.S.	27.6 x 10^6	21,300	66,200
SUPPORT MATERIAL DATA (200°F)			
SA240, Type 304L (upper part of support feet)	27.6 x 10^6	21,300	66,200
SA-564-630 (lower part of support feet; age hardened at 1100°F)	27.6 x 10^6	106,300	140,000

Table 6.4.1 TIME-HISTORY STATISTICAL CORRELATION RESULTS	
OBE	
Data1 to Data2	0.090
Data1 to Data3	0.016
Data2 to Data3	0.008
SSE	
Data1 to Data2	0.118
Data1 to Data3	-0.021
Data2 to Data3	-0.127

Data1 corresponds to the time-history acceleration values along the **X axis (North)**

Data2 corresponds to the time-history acceleration values along the **Y axis (West)**

Data3 corresponds to the time-history acceleration values along the **Z axis (Vertical)**

Table 6.5.1

Degrees-of-freedom

LOCATION (Node)	DISPLACEMENT			ROTATION		
	U_x	U_y	U_z	θ_x	θ_y	θ_z
1	p_1	p_2	p_3	q_4	q_5	q_6
2	p_7	p_8	p_9	q_{10}	q_{11}	q_{12}

Node 1 is assumed to be attached to the rack at the bottom most point.
Node 2 is assumed to be attached to the rack at the top most point.
Refer to Figure 6.5.1 for node identification.

LOCATION (Node)	U_x	U_y
2*	p_{13}	p_{14}
3*	p_{15}	p_{16}
4*	p_{17}	p_{18}
5*	p_{19}	p_{20}
1*	p_{21}	p_{22}

where the relative displacement variables q_i are defined as:

$p_i = q_i(t) + U_x(t) \quad i = 1,7,13,15,17,19,21$

$= q_i(t) + U_y(t) \quad i = 2,8,14,16,18,20,22$

$= q_i(t) + U_z(t) \quad i = 3,9$

$= q_i(t) \quad i = 4,5,6,10,11,12$

p_i denotes absolute displacement (or rotation) with respect to inertial space

q_i denotes relative displacement (or rotation) with respect to the floor slab

* denotes fuel mass nodes

U(t) are the three known earthquake displacements

Table 6.5.2 (DYNARACK) NUMBERING SYSTEM FOR GAP ELEMENTS AND FRICTION ELEMENTS		
I. Nonlinear Springs (Type 3 Gap Elements - 520 Total)		
Number	**Node Loc.**	**Description**
1	Support S1	Z compression-only element
2	Support S2	Z compression-only element
3	Support S3	Z compression-only element
4	Support S4	Z compression-only element
5	2,2*	X rack/fuel assembly impact element between nodes 2 and 2*
6	2,2*	X rack/fuel assembly impact element between nodes 2 and 2*
7	2,2*	Y rack/fuel assembly impact element between nodes 2 and 2*
8	2,2*	Y rack/fuel assembly impact element between nodes 2 and 2*
9-360	Impact elements corresponding to the rattling masses at nodes 1*, 3*, 4* and 5* (similar to elements 5 thru 8)	
361-520	Bottom and Top Cross section of Rack (around edge)	Inter-rack impact elements

Table 6.5.2

(DYNARACK) NUMBERING SYSTEM FOR GAP ELEMENTS AND FRICTION ELEMENTS

II. Linear Springs (Type 1 Elements - 90 Total)

Number	Rack No.	Description
1	1	Rack beam bending element (x-z plane)
2	1	Rack shear deformation element (x-z plane)
3	1	Rack beam bending element (y-z plane)
4	1	Rack shear deformation element (y-z plane)
5	1	Rack beam axial deformation element
6	1	Rack beam torsional deformation element
7-12	2	Similar to elements 1 thru 6
13-18	3	Similar to elements 1 thru 6, continue to Rack 15

III. Piece-wise Linear Friction Springs (Type 2 Elements - 120 Total)

Number	Rack No.	Description
1	1	Pedestal 1, X direction
2	1	Pedestal 1, Y direction
3	1	Pedestal 2, X direction
4	1	Pedestal 2, Y direction
5	1	Pedestal 3, X direction
6	1	Pedestal 3, Y direction
7	1	Pedestal 4, X direction
8	1	Pedestal 4, Y direction
9-16	2	Similar to elements 1 thru 8
17-24	3	Similar to elements 1 thru 8, continue to Rack 15

Table 6.9.1

COMPARISON OF BOUNDING CALCULATED LOADS/STRESSES VS/ CODE ALLOWABLES AT IMPACT LOCATIONS AND WELDS

Item/Location	Calculated	Allowable
Fuel assembly/cell wall impact, lbf.	802	3,187*
Rack Cell to baseplate weld, psi	11,520 (SSE)	35,748 (SSE)
Female pedestal to baseplate weld, psi	6,975 (SSE) 4,194 (OBE)	35,748 (SSE) 19,860 (OBE)
Cell to cell welds, lbf.	2,268**	7,796

* Based on the limit load for a cell wall. The allowable load on the fuel assembly itself may be less than this value but is greater than 802 lbs

**Based on the fuel assembly to cell wall impact load simultaneously applied in two orthogonal directions.

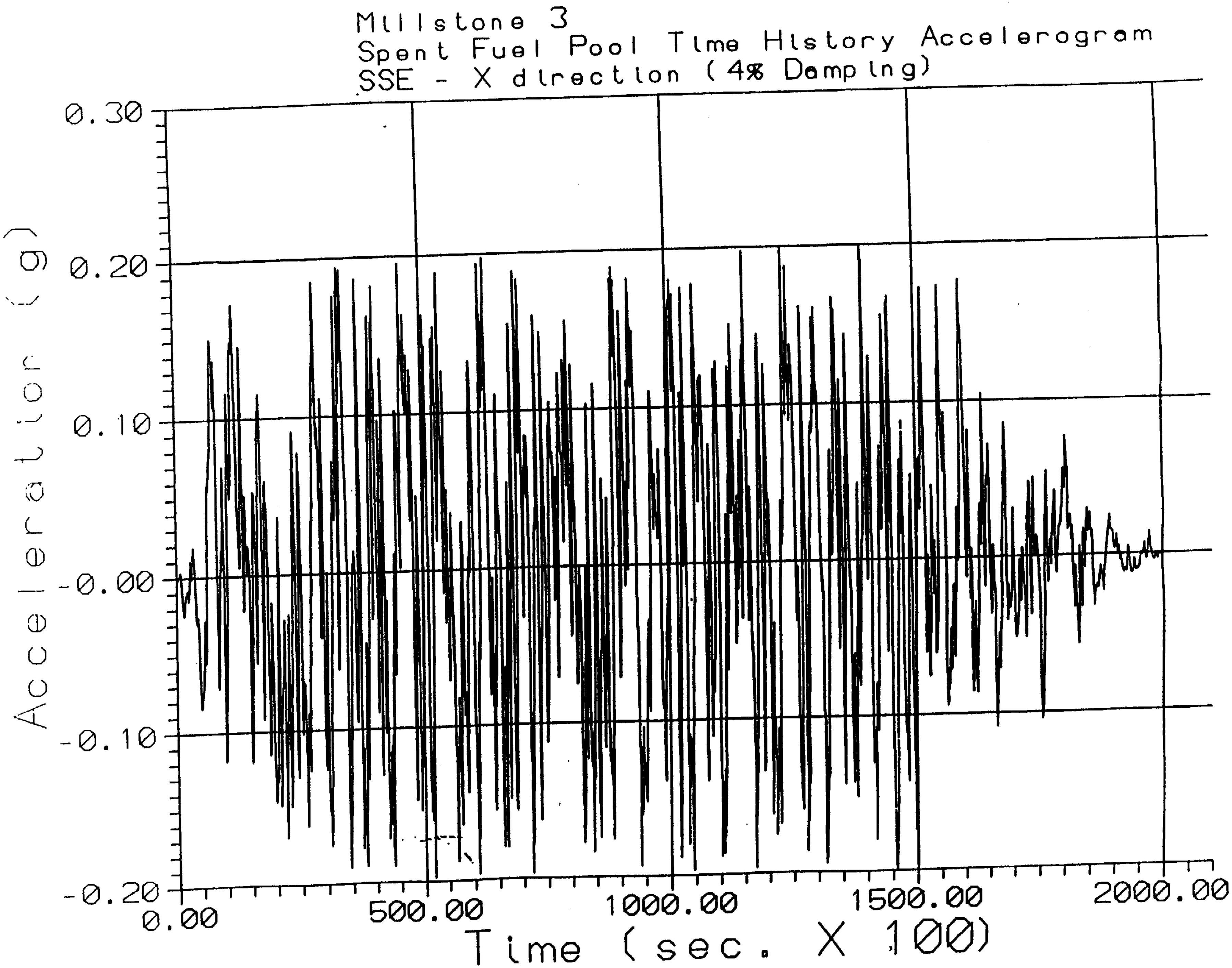
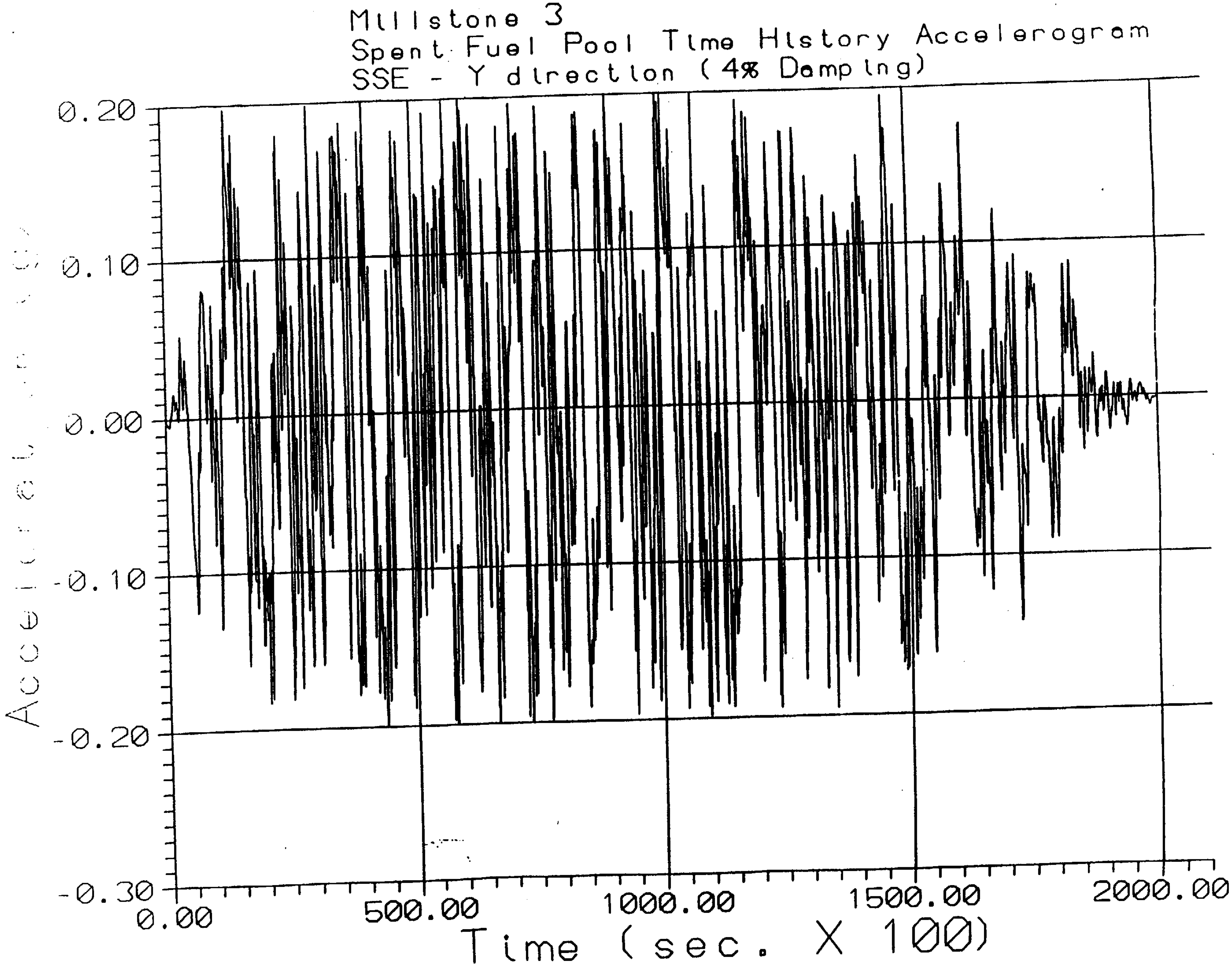
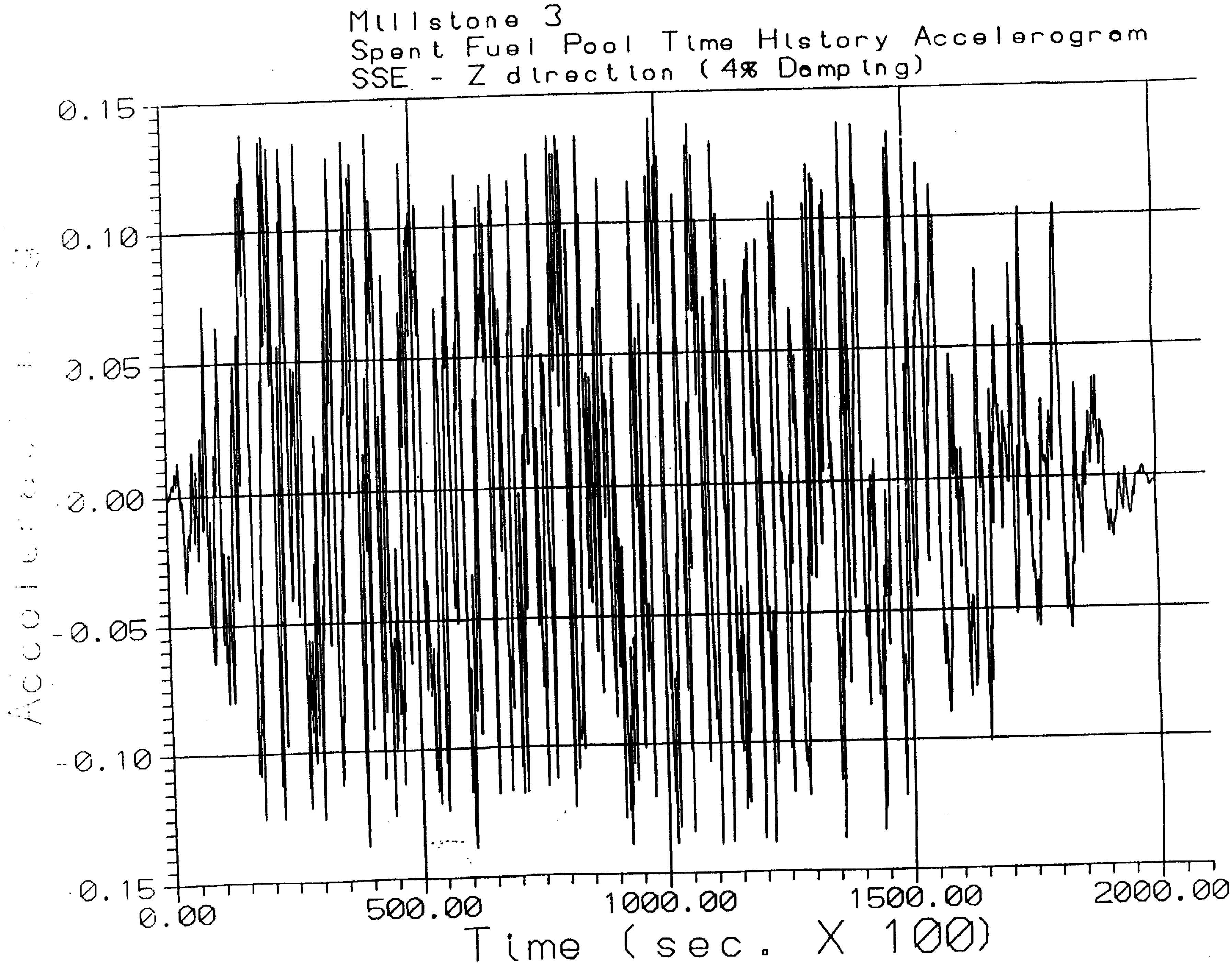
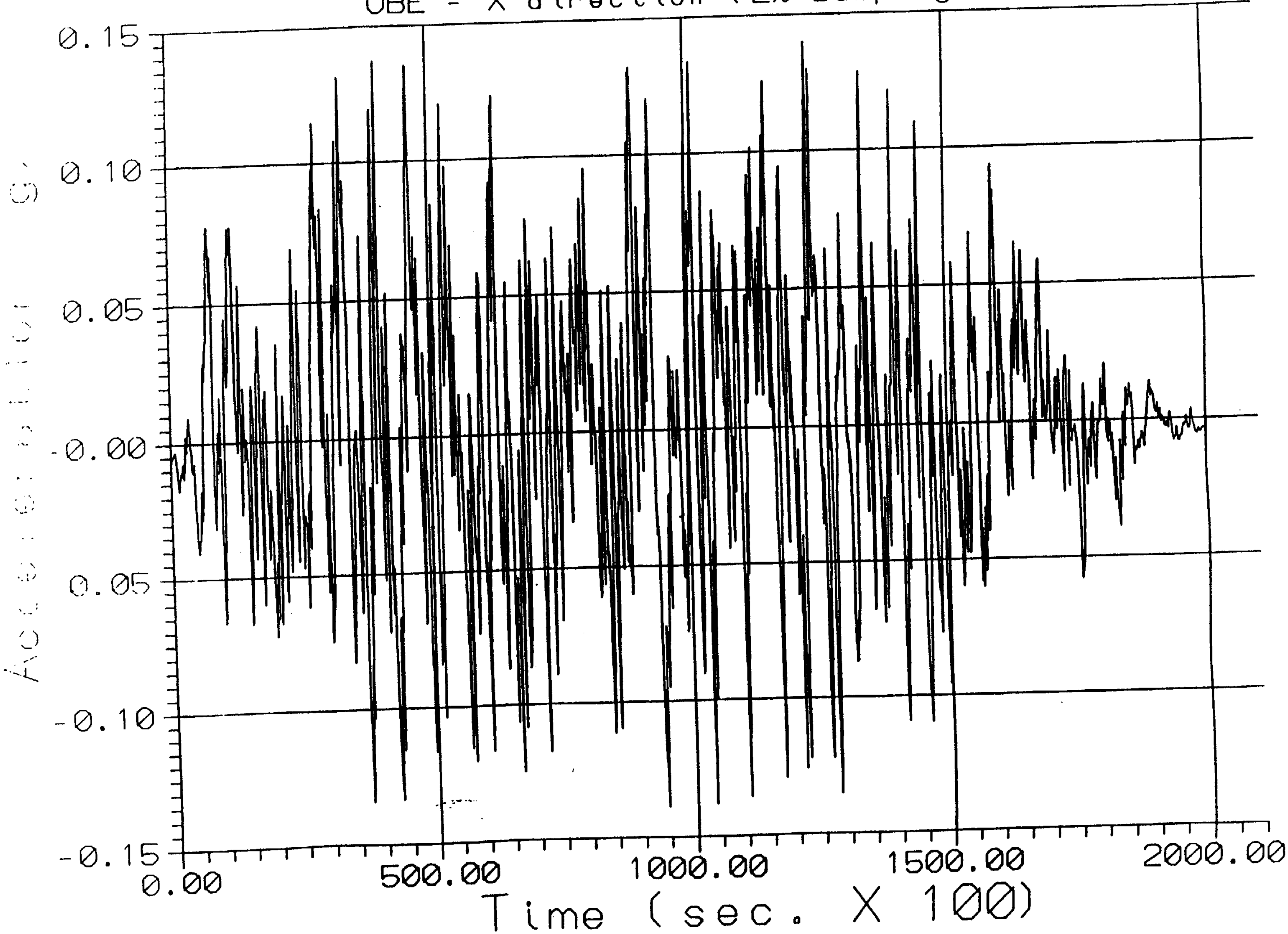

Millstone 3
Spent Fuel Pool Time History Accelerogram
SSE - X direction (4% Damping)
0.30
0.20
0.10
-0.00
-0.10
-0.20
Acceleration (g)
0.00
500.00
1000.00
1500.00
2000.00
Time (sec. X 100)

FIG 6.4.2 Holtec Project No. 4




Millstone 3
Spent Fuel Pool Time History Accelerogram
SSE - Z direction (4% Damping)
0.15
0.10
0.05
0.00
-0.05
-0.10
-0.15
0.00
500.00
1000.00
1500.00
2000.00
Time (sec. X 100)
Acc

Millstone 3
Spent Fuel Pool Time History Accelerogram
OBE - X direction (2% Damping)



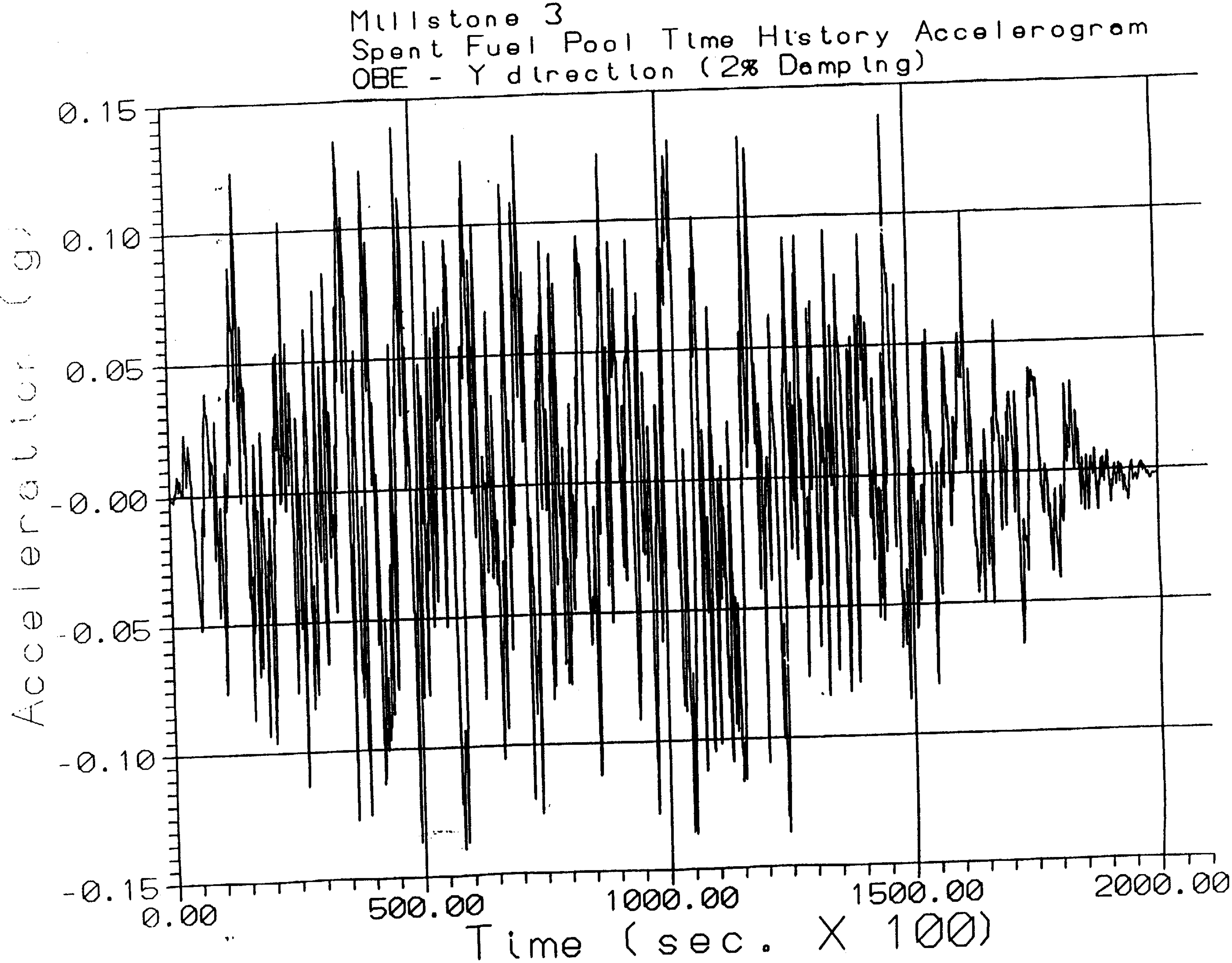
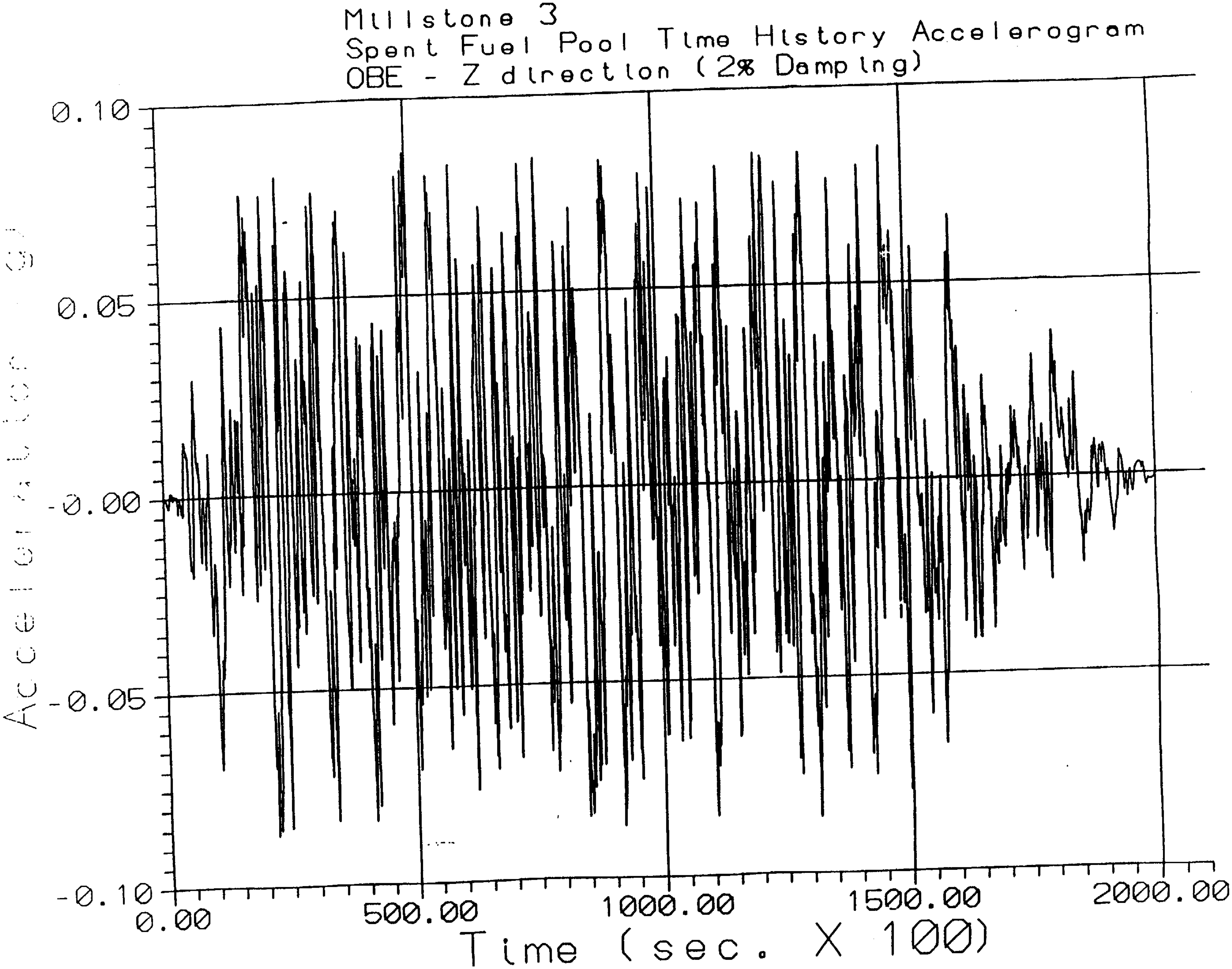

Millstone 3
Spent Fuel Pool Time History Accelerogram
OBE - Y direction (2% Damping)
0.15
0.10
0.05
-0.00
-0.05
-0.10
-0.15
Acceleration (g)
0.00
500.00
1000.00
1500.00
2000.00
Time (sec. X 100)

FIGURE (.6

Holtec Project No. 711



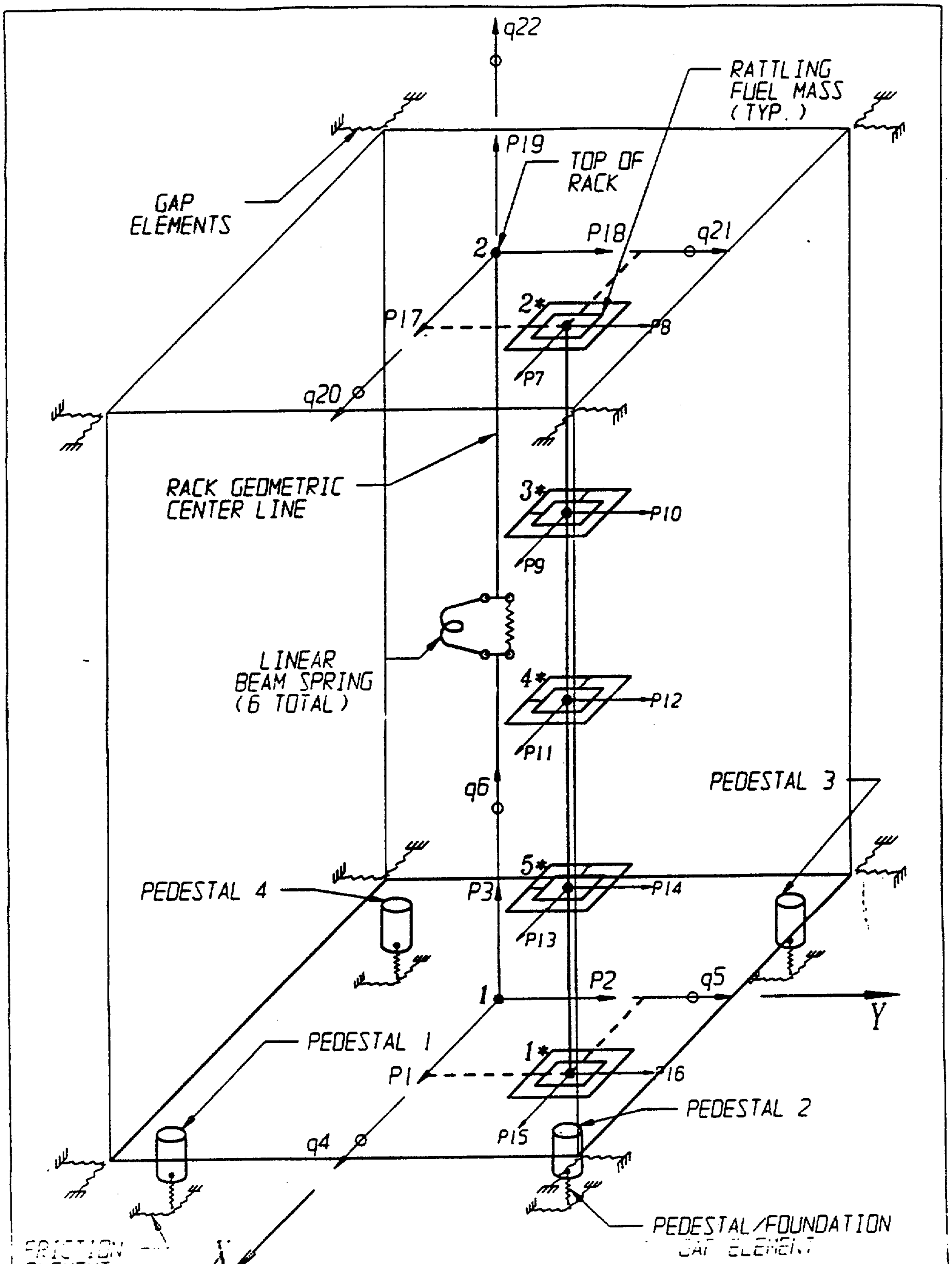


FIGURE 6.5.1; SCHEMATIC OF THE DYNAMIC MODEL OF A SINGLE RACK MODULE USED IN DYNARACK

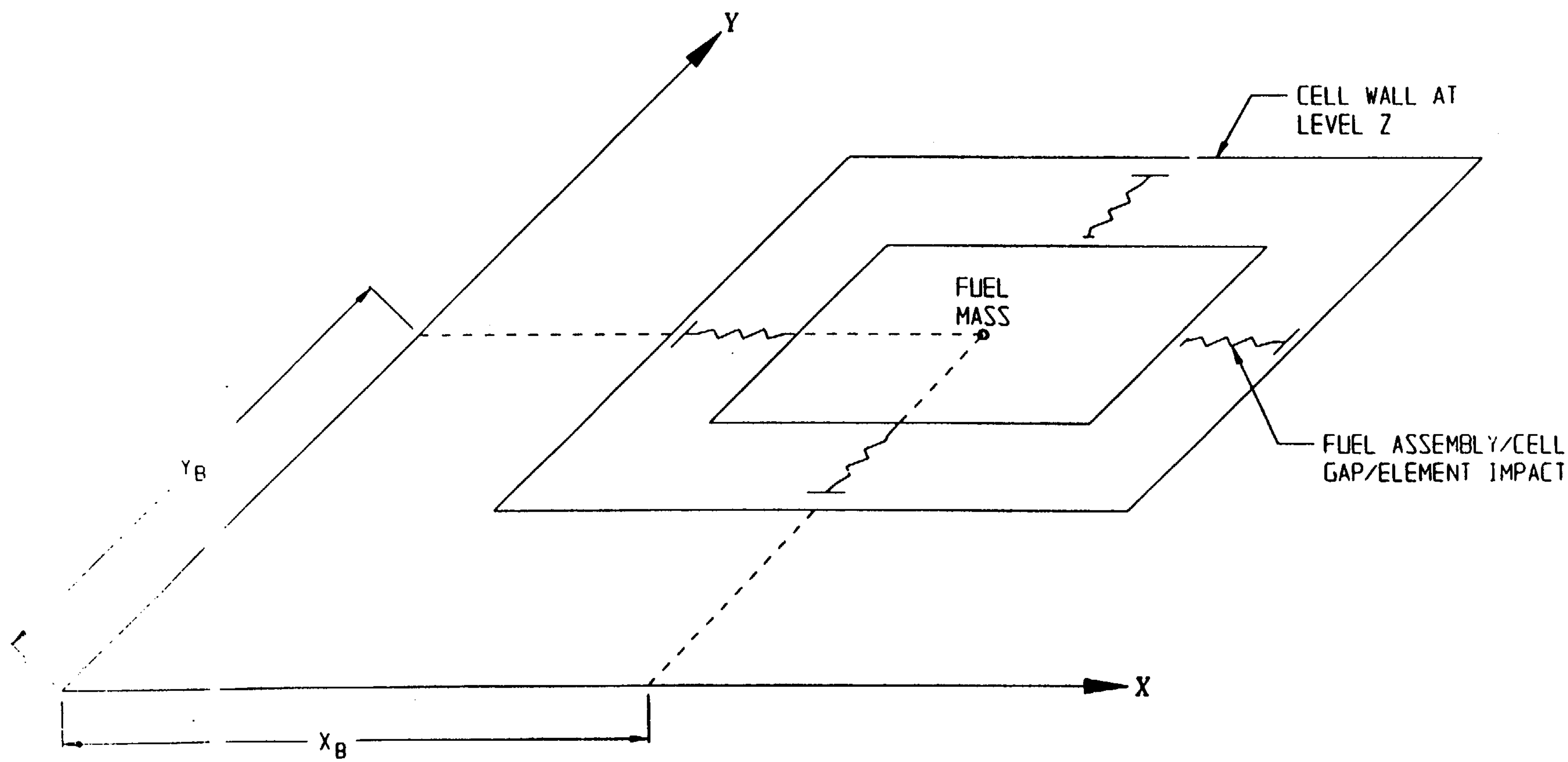


FIGURE 6.5.2 ; FUEL-TO-RACK GAP/IMPACT ELEMENTS AT LEVEL OF RATTLING MASS

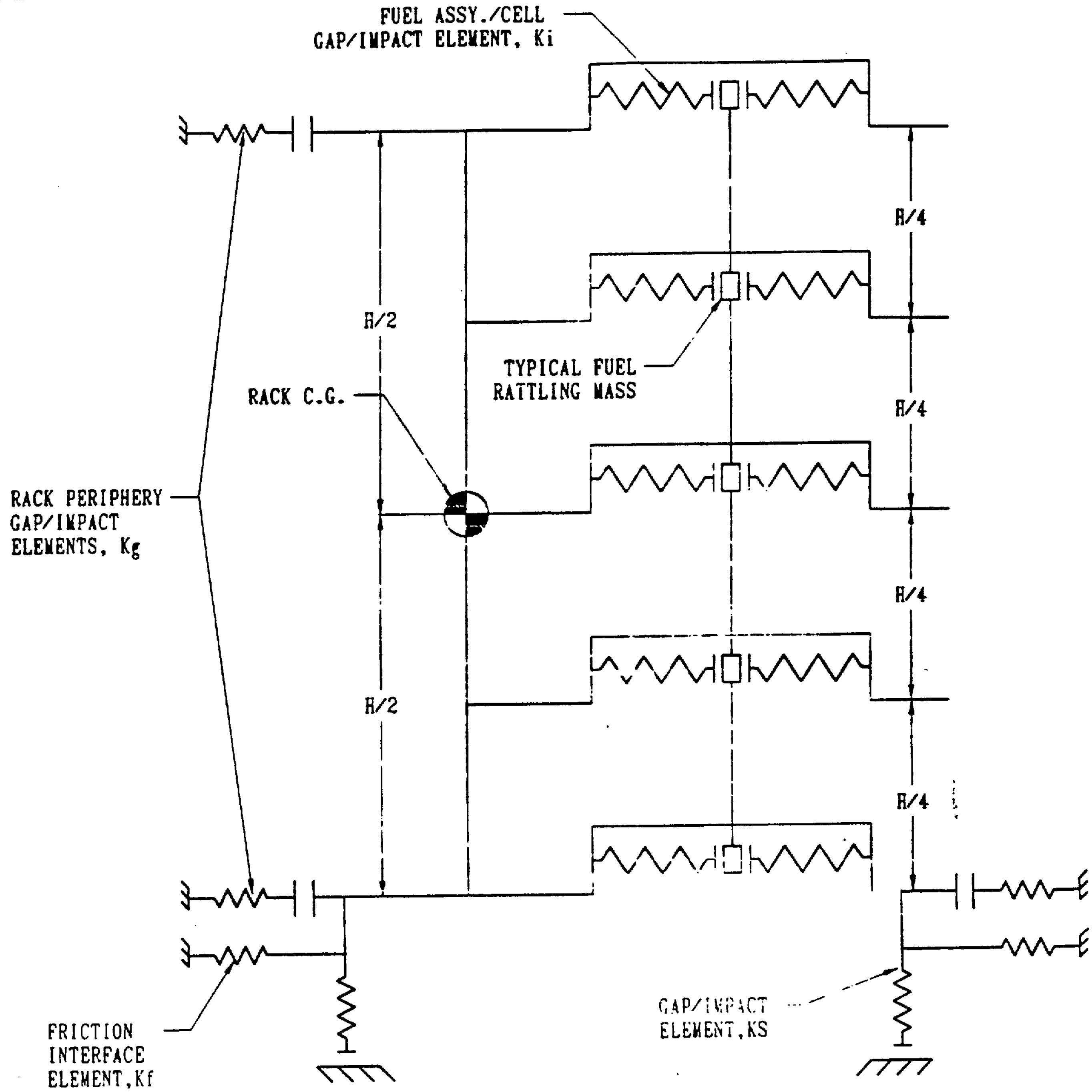
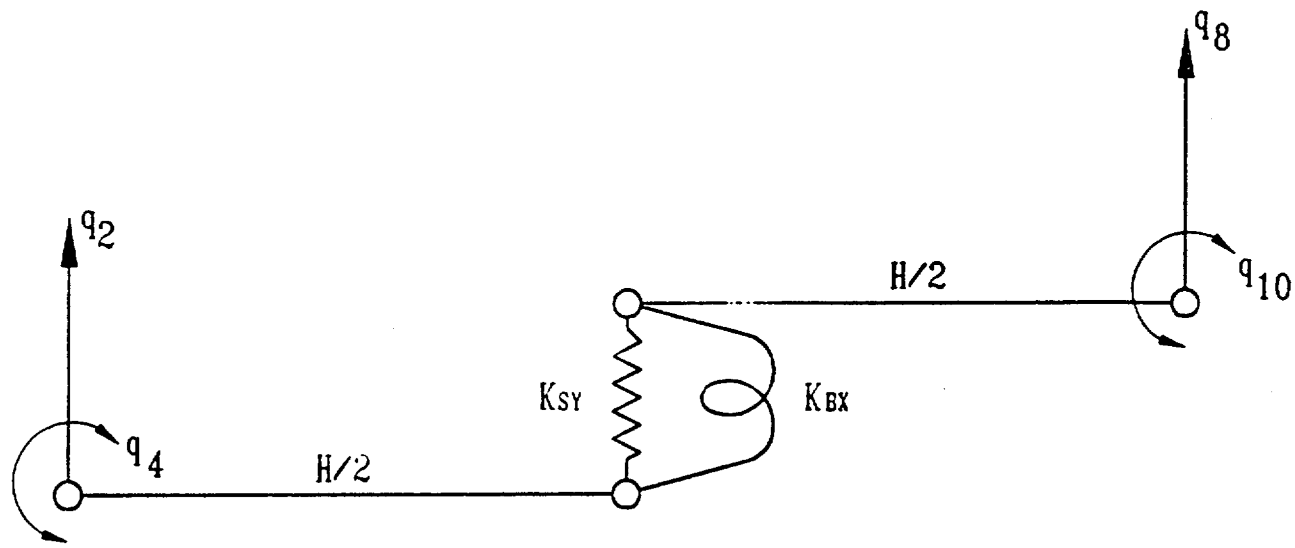


FIGURE 6.5.3; TWO DIMENSIONAL VIEW OF THE SPRING-MASS SIMULATION



RACK DEGREES-OF-FREEDOM FOR Y-Z PLANE BENDING
WITH SHEAR AND BENDING SPRING

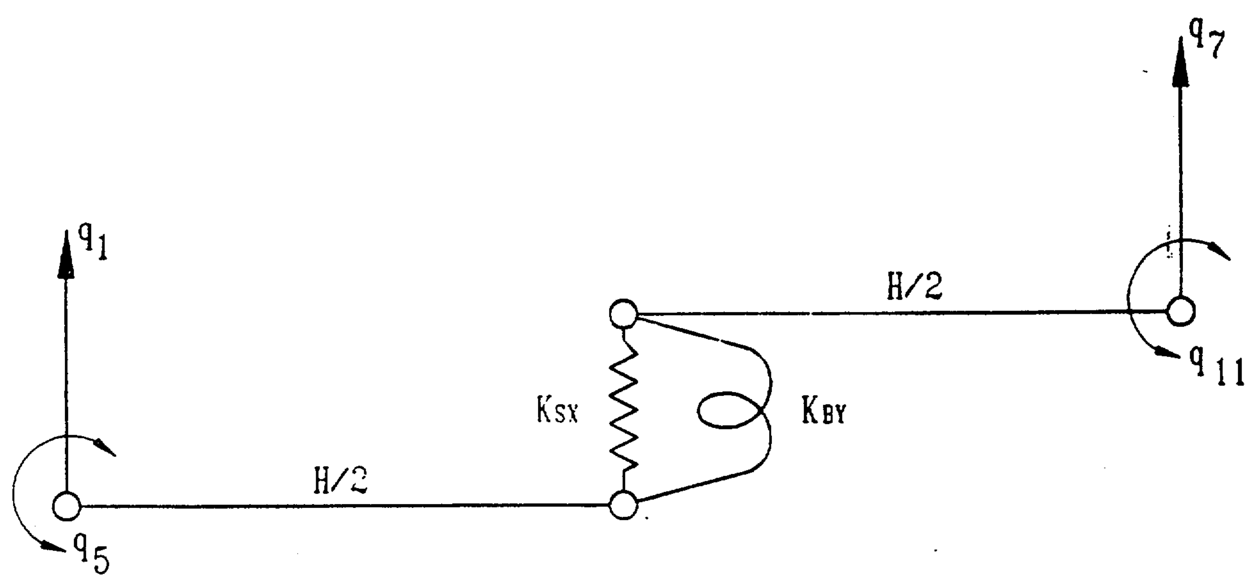


FIGURE 6.5.4; RACK DEGREES-OF-FREEDOM FOR X-Z PLANE
BENDING WITH SHEAR AND BENDING SPRING

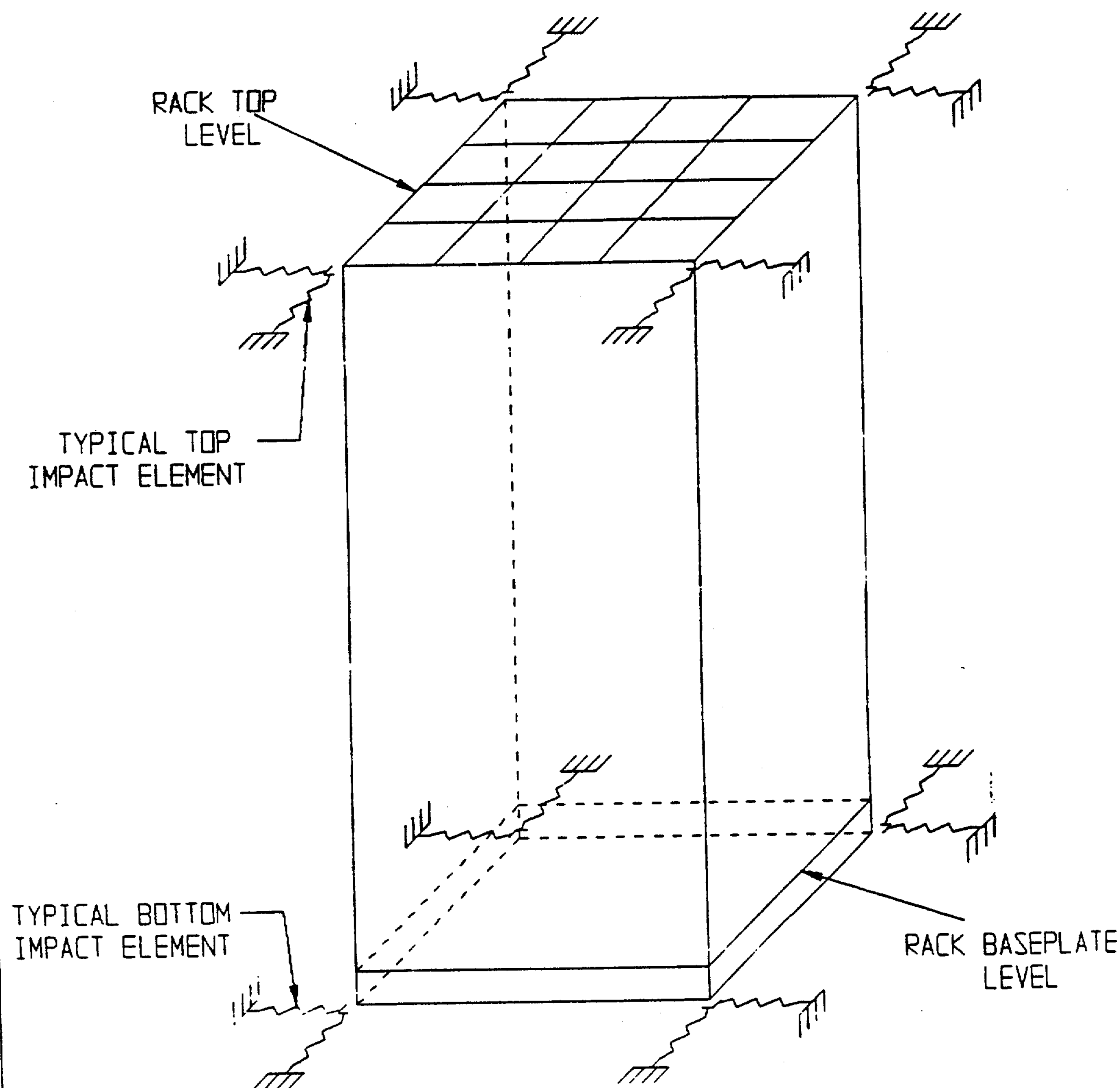


FIGURE 6.5.5 RACK-TO-RACK IMPACT SPRINGS

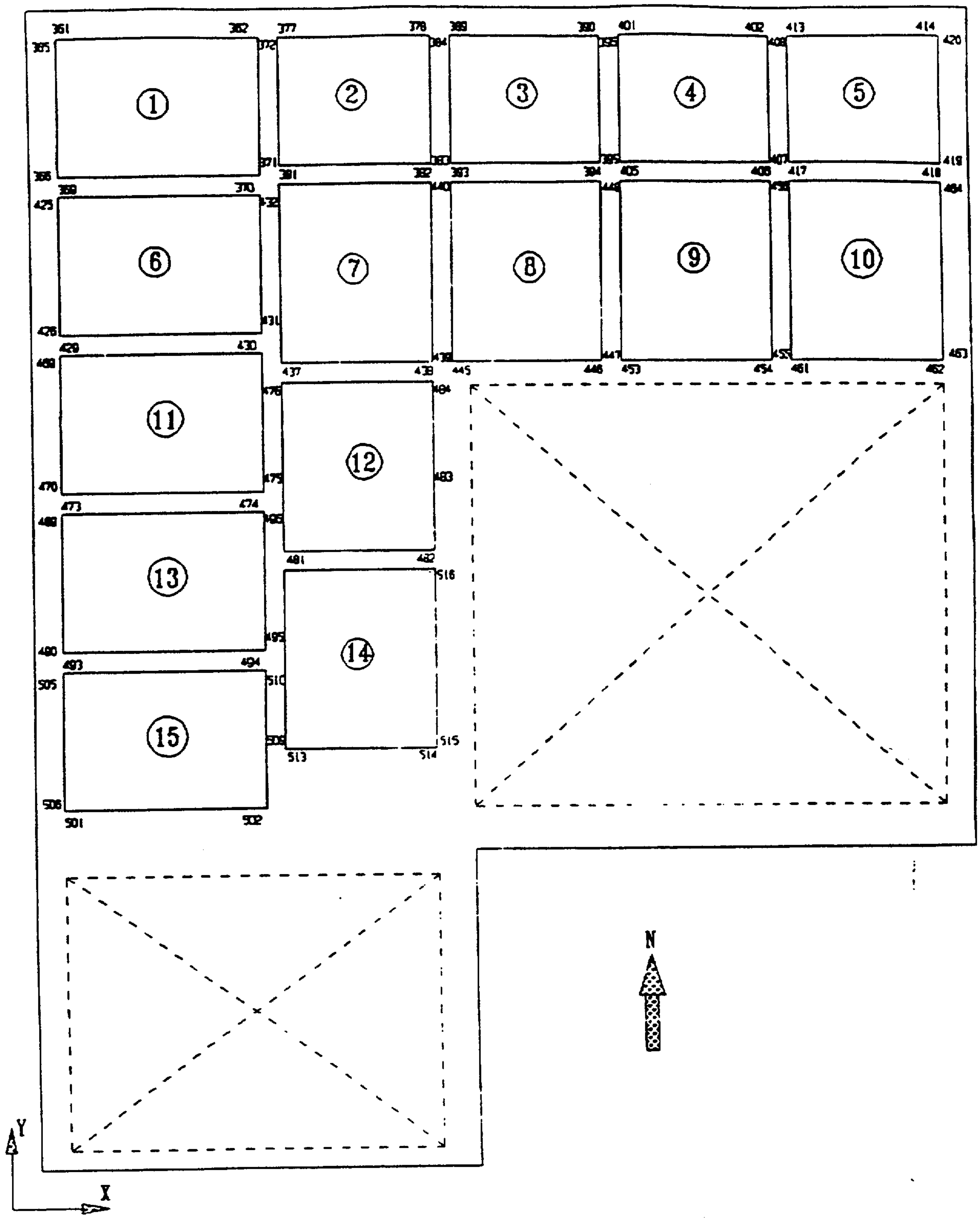


FIGURE 6.8.1:GAP SPRING NUMBERING SCHEME (BOTTOM)

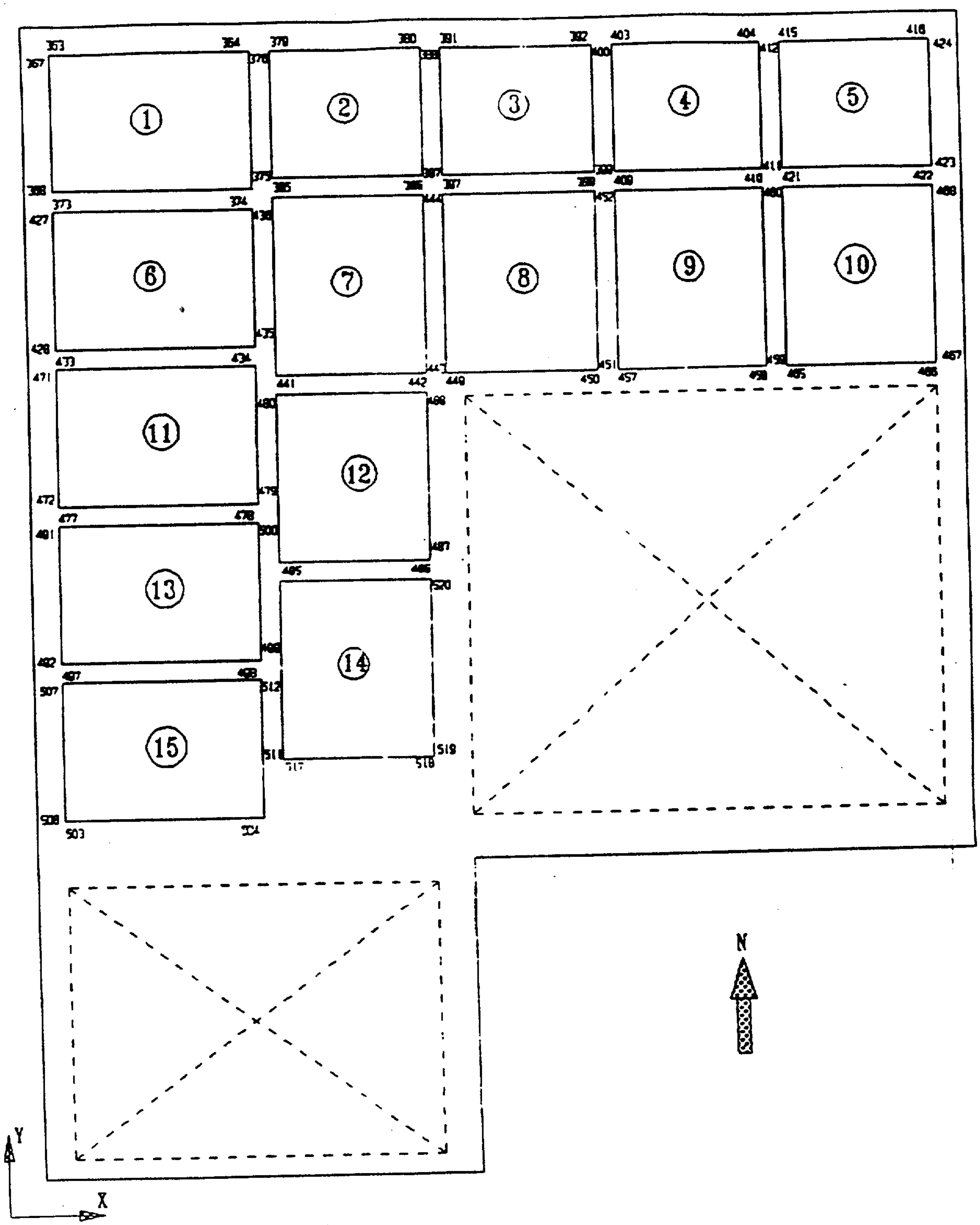
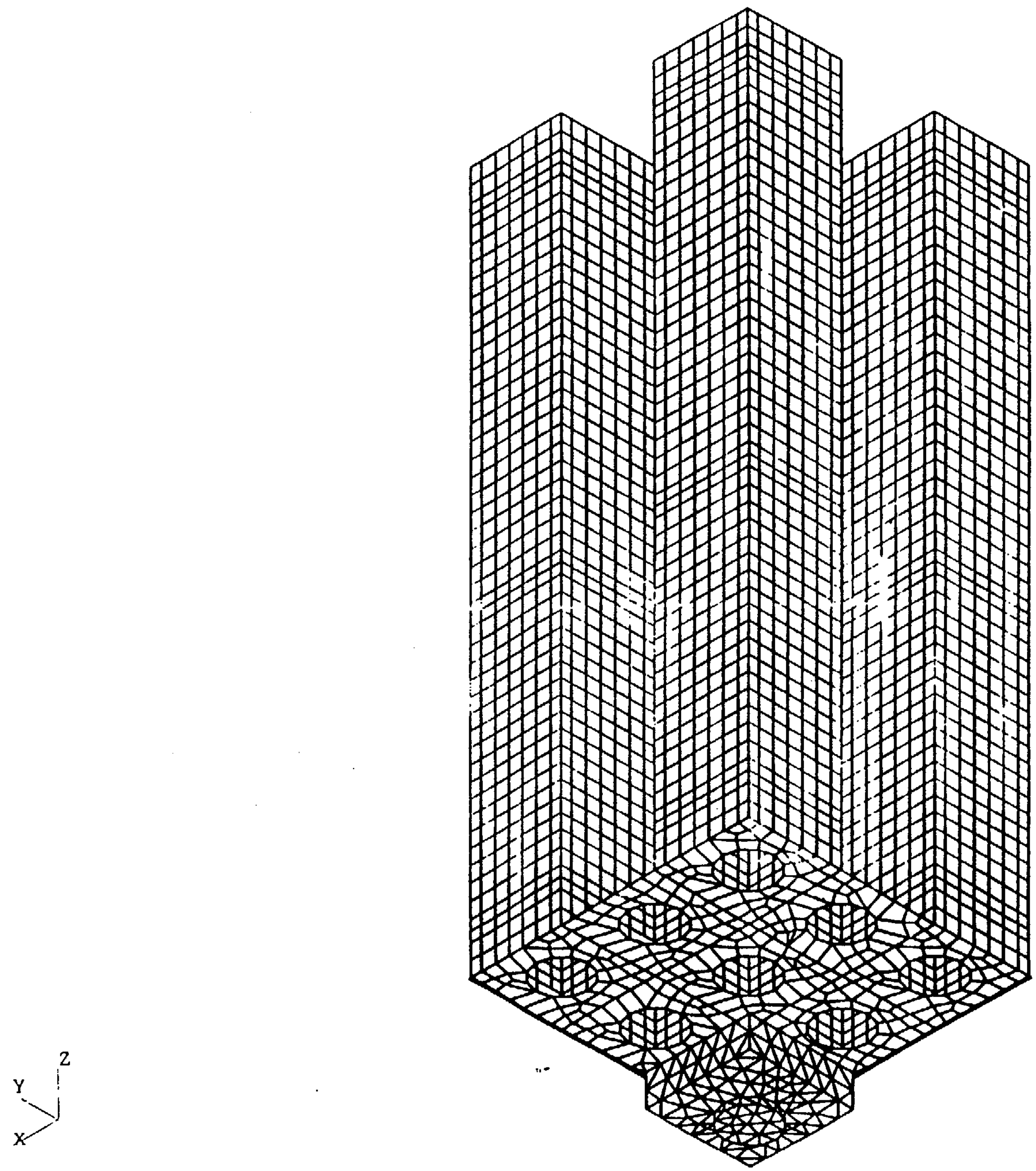


FIGURE 6.8.2; GAP SPRING NUMBERING SCHEME (TOP)



Overall Finite Elemet Model

Figure 6.10.1

7.0 FUEL HANDLING AND CONSTRUCTION ACCIDENTS

7.1 Introduction

The USNRC OT position paper [7.1] specifies that the design of the rack must ensure the functional integrity of the spent fuel racks under all credible drop events in the spent fuel pool. This section contains synopses of the analyses carried out to demonstrate the regulatory compliance of the proposed racks under postulated fuel assembly drop scenarios germane to MP3.

Two scenarios are postulated in addition to the fuel assembly drop: 1) dropping the heaviest rack in the pool from the maximum possible height during rack installation and 2) dropping a pool gate onto the racks. The first accident requires showing that the pool structure can withstand a rack drop so as to prevent a rapid loss of water. The gate scenario requires showing that the drop will not damage either the fuel assemblies themselves or the poison material on the racks.

7.2 Description of Fuel Handling Accidents

In the evaluation of fuel handling accidents, the concern is with the damage to the storage racks. The configuration of the fuel assemblies, rack cell size, spacing, and neutron absorber material must remain consistent with the configurations used in the criticality evaluations. Maintaining these designed configurations will ensure that the results of the criticality evaluations remain valid.

Radiological concerns due to fuel damage are not an issue, since the fuel handling design basis accident considers the worst case condition of a falling assembly, which remains unchanged. This condition is a fuel assembly falling onto another assembly. Fuel damage subsequent to a fuel assembly drop is primarily influenced by the weight and design of the fuel assembly, the drop height (which determines the kinetic energy upon impact), and the orientation of the falling assembly. Since none of these parameters are changed under the proposed modification, the number of fuel rods damaged during a fuel assembly drop remains consistent with the previously analyzed fuel handling design basis accident.

During the previously evaluated design basis event the kinetic energy of the falling assembly is maximized by selection of the greatest drop distance (from the Handling Machine to the floor of the pool). A drop event considering a falling assembly striking the top of the storage cell represents a significantly reduced drop height and a corresponding reduction in the kinetic energy of the falling assembly. The new storage configuration does not change the elevation of the top of stored fuel. Therefore, the new configuration does not represent a significant change in the kinetic energy of a fuel assembly directly striking the top of stored fuel. A falling fuel assembly striking the top of the racks and causing sufficient deformation to also strike the top of a stored assembly is also possible, but is even less limiting. The falling assembly would impart far less kinetic energy to the stored assembly than a direct impact, since a significant portion of the kinetic energy of the falling assembly would be absorbed by damage to the racks. Therefore, the radiological consequences resulting from a fuel drop accident continue to be bounded by the previously evaluated design basis accident.

Two categories of fuel assembly accidental drop events are considered. In the so-called "shallow drop" event, a fuel assembly, along with the portion of handling tool which is severable in the case of a single element failure, is assumed to drop vertically and hit the top of the rack. Inasmuch as the new racks are of honeycomb construction, the deformation produced by the impact is expected to be confined to the region of collision. However, the "depth" of damage to the affected cell walls must be demonstrated to remain limited to the portion of the cell above the top of the "active fuel region", which is essentially the elevation of the top of the Boral neutron absorber. To meet this criterion, the plastic deformation of the rack cell wall should not extend more than 18.125 inches (downwards) from the top of the rack. This will ensure that the configurations considered in the criticality evaluations are not compromised.

In order to utilize an upper bound of kinetic energy at impact, the impactor is assumed to weigh 2,100 lbs and the free-fall height is assumed to be 36 inches through air; resulting in 75,600 lbs-in of kinetic energy. The impactor weight corresponds to the weight of a fuel assembly along with a Rod Control Cluster Assembly (RCCA) and the fuel handling tool. This weight was chosen to bound the drop energies that result from a 2,200 lb buoyant weight (MP3 Tech Spec) dropped from

30 inches. This results in an impact energy of 66,000 lbs-in. Also, the analyzed fuel drop bounds the analyzed fuel drop on Westinghouse racks (2,112 lbs. buoyant weight dropped 30 inches). This results in an impact energy of 63,360 lbs.-in. During normal fuel handling, a fuel assembly cannot reach a height greater than 30 inches above the racks. Therefore, the case considered here envelopes the existing design basis.

It is readily apparent from the description of the rack modules in Section 3 that the impact resistance of a rack at its periphery is less than its interior. Accordingly, the potential shallow drop scenario is postulated to occur at a rack periphery cell in the manner shown in Figure 7.2.1.

In order to maximize the penetration into the top of the rack by the falling assembly, the rack is considered empty (i.e., without assemblies or RCCAs). Exclusion of the stored fuel from the model eliminates the possibility of sharing the kinetic energy with the rack, thus maximizing rack damage (e.g., depth of penetration).

Finally, the fuel assembly is assumed to hit the rack in a manner to inflict maximum damage. The impact zone is chosen to minimize the cross sectional area which experiences the deformation. Figure 7.2.2 depicts the impacted rack in plan view.

The second class of "fuel drop event" postulates that the impactor falls through an empty storage cell impacting the rack baseplate. This so-called "deep drop" scenario threatens the structural integrity of the "baseplate". If the baseplate is pierced, then the fuel assembly might damage the pool liner and/or create an abnormal condition of the enriched zone of fuel assembly outside the "poisoned" space of the fuel rack. To preclude damage to the pool liner, and to avoid the potential of an abnormal fuel storage configuration in the aftermath of a deep drop event, it is required that the baseplate remain unpierced and that the maximum lowering of the fuel assembly support surface is less than the distance from the bottom of the rack baseplate to the liner.

The deep drop event can be classified into two scenarios, namely, drop through cell located above a support leg (Figure 7.2.3), and drop in an interior cell away from the support pedestal (Figure 7.2.4).

In the former deep drop scenario (Figure 7.2.3), the baseplate is buttressed by the support pedestal and presents a hardened impact surface, resulting in a high impact load. The principal design objective is to ensure that the support pedestal does not pierce the lined, reinforced concrete pool slab.

The baseplate is not quite as stiff at cell locations away from the support pedestal (Figure 7.2.4). Baseplate severing and large deflection of the baseplate (such that the liner would be impacted) would constitute an unacceptable result.

7.3 Mathematical Model

In the first step of the solution process, the velocity of the dropped object (impactor) is computed for the condition of underwater free fall. Table 7.3.1 contains the results for the three drop events.

In the second step of the solution, an elasto-plastic finite element model of the impacted region on Holtec's computer Code PLASTIPACT (Los Alamos National Laboratory's DYNA3D implemented on Holtec's QA system) is prepared. PLASTIPACT simulates the transient collision event with full consideration of plastic, large deformation, wave propagation, and elastic/plastic buckling modes. For conservatism, the impactor in all cases is conservatively assumed to be *rigid*. The physical properties of material types undergoing deformation in the postulated impact events are summarized in Table 7.3.2.

7.4 Results

7.4.1 Shallow Drop Events

Figure 7.4.1 provides an isometric of the finite element model utilized in the shallow drop impact analysis.

Dynamic analyses show that the top of the impacted region undergoes severe localized deformation. Figure 7.4.2 shows an isometric view of the post-impact geometry of the rack for the shallow drop scenario. The maximum depth of plastic deformation is limited to 6.64 inches, which is below the design limit of 18.125 inches. Figure 7.4.3 shows the plan view of the post-collision geometry. Approximately 10% of the cell opening in the impacted cell is blocked.

7.4.2 Deep Drop Events

The deep drop scenario (Figure 7.4.4) wherein the impact region is located above the support pedestal (Figure 7.4.4a) is found to produce a negligible deformation on the baseplate. The maximum Von Mises stress occurs in a localized region at is limited to only 25 ksi. Insignificant plastic strain occurs in the liner. Therefore, it is concluded that the pool liner will not be damaged.

The deep drop condition through an interior cell (Figure 7.4.4b) does produce some deformation of the baseplate and localized severing of the baseplate/cell wall welds (Figure 7.4.5). However, the fuel assembly support surface is lowered by a maximum of 2.9 inches, which is less than the distance of 4-5/8 inches from the baseplate to the liner. Therefore, the pool liner will not be damaged.

7.5 Rack Drop

The drop of a rack during the reracking process was also postulated. This evaluation considered a rack to be dropped to the bottom of the pool from a height of 40 feet. The analysis of damage to the liner and underlying concrete was determined by neglecting any bearing pads at the impact site and considering that the pedestal directly strikes the unprotected liner. It was determined that the pool floor would not suffer structural damage.

7.6 Gate Drop

A drop of the spent fuel pool canal gate was also analyzed. The analysis considered a drop of the

5000 lb (dry weight) from a conservatively assumed height of 36" onto empty spent fuel storage racks and onto the spent fuel pool liner. The actual gate carrying height above the racks is 21". The MP3 technical specifications prohibit the gate from travel over spent fuel.

The existing Westinghouse racks and the new Holtec racks were evaluated. The Region 2 type Holtec rack was selected for the drop evaluation since they contain less material and weld connections than the Region 1 type racks. In both cases a peripheral row of cells was chosen to ensure that the gate transmitted its entire energy into the racks most vulnerable location.

The results demonstrate that a potential gate drop would penetrate the rack cell for a distance of 5 inches for the Westinghouse racks and 7.45 inches for the Holtec racks, causing local damage and deformation. However the damage is limited to the upper cellular region of the rack and does not extend to the rack cells in the active fuel zone. The drop would also not damage the poison material (Boral) in the Holtec racks. The racks would therefore remain functional with respect to storage of spent fuel in cells adjacent to those potentially impacted by the gate drop. It was also shown that the gate would not pierce the spent fuel pool liner.

At this time NU will not license to allow fuel to be under the safe load path of a gate during gate movement. It should be noted that the gate drop issues do not need to be addressed until the new racks are installed. The Canal gate is not located close to the existing racks.

7.7 Closure

The fuel assembly drop accident events postulated for the pools were analyzed and found to produce localized damage well within the design limits for the racks. The configuration of the fuel and poison (Boral) is not compromised from the configurations analyzed in the criticality evaluations discussed in Section 4.0. Therefore, there are no criticality concerns for these accidents. A construction accident event wherein the heaviest rack falls from a 40' height onto the pool floor was also considered. Analyses show that the pool structure will not suffer structural damage.

7.8 References

[7.1] "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

TABLE 7.3.1

IMPACT EVENT DATA

Case	Impactor Weight (lbs)	Impactor	Drop Height (inches)	Impact Velocity (inch/sec)
1. Shallow drop event	2,100	Fuel Assembly	36	155
2. Deep drop event	2,100	Fuel Assembly	204.375	355
3. Construction event	Heaviest Rack	Rack Module	480	300
4. Gate Drop	5,000	Gate	36	144

Table 7.3.2

Material Definition

Material Name	Type	Density (pcf)	Elastic Modulus (psi)	Stress (psi)		Strain	
				First Yield	Failure	Elastic	Failure
Stainless Steel	SA240-304L	490	2.760e+07	2.130e+04	6.620e+04	7.717e-04	3.800e-01
Stainless Steel	SA240-304	490	2.760e+07	2.500e+04	7.100e+04	7.717e-04	3.800e-01
Stainless Steel	SA564-630	490	2.760e+07	1.063e+05	1.400e+05	3.851e-02	3.800e-01
Concrete	4000	150	3.605e+06	4.000e+03	2.022e+04	1.110e-03	5.500e-02

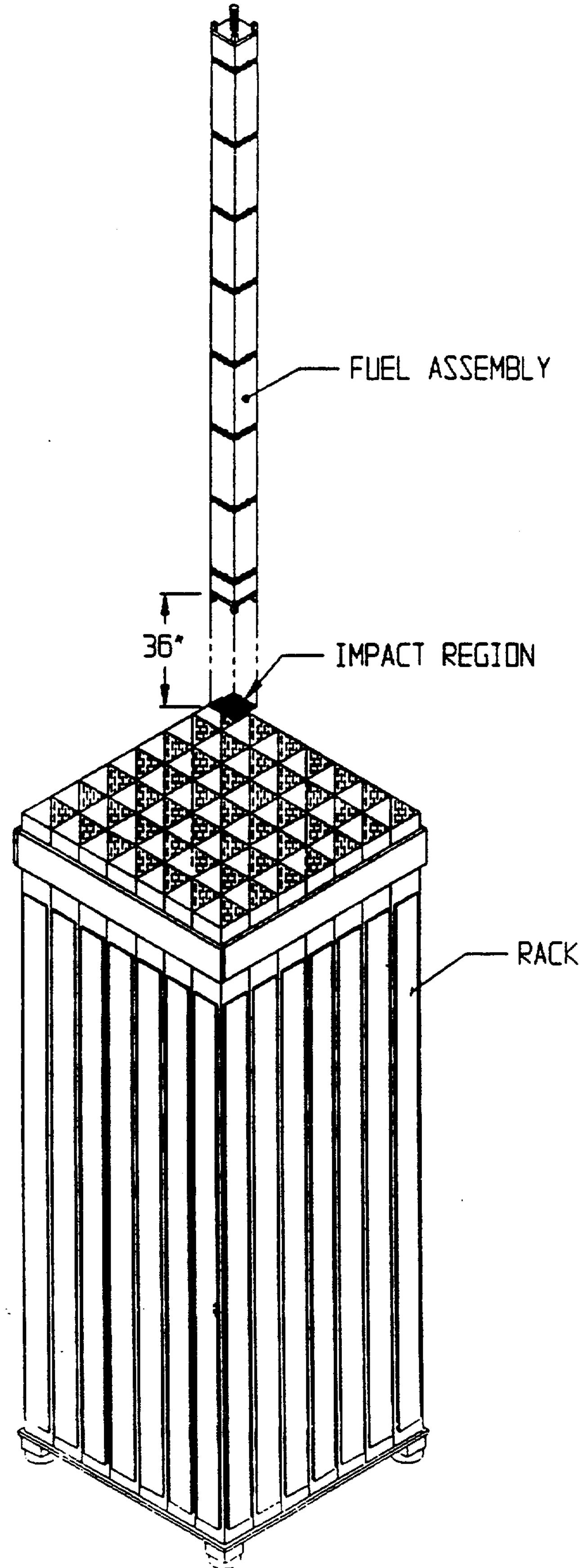


FIGURE 7.2.1; SHALLOW DROP ON A PERIPHERAL CELL

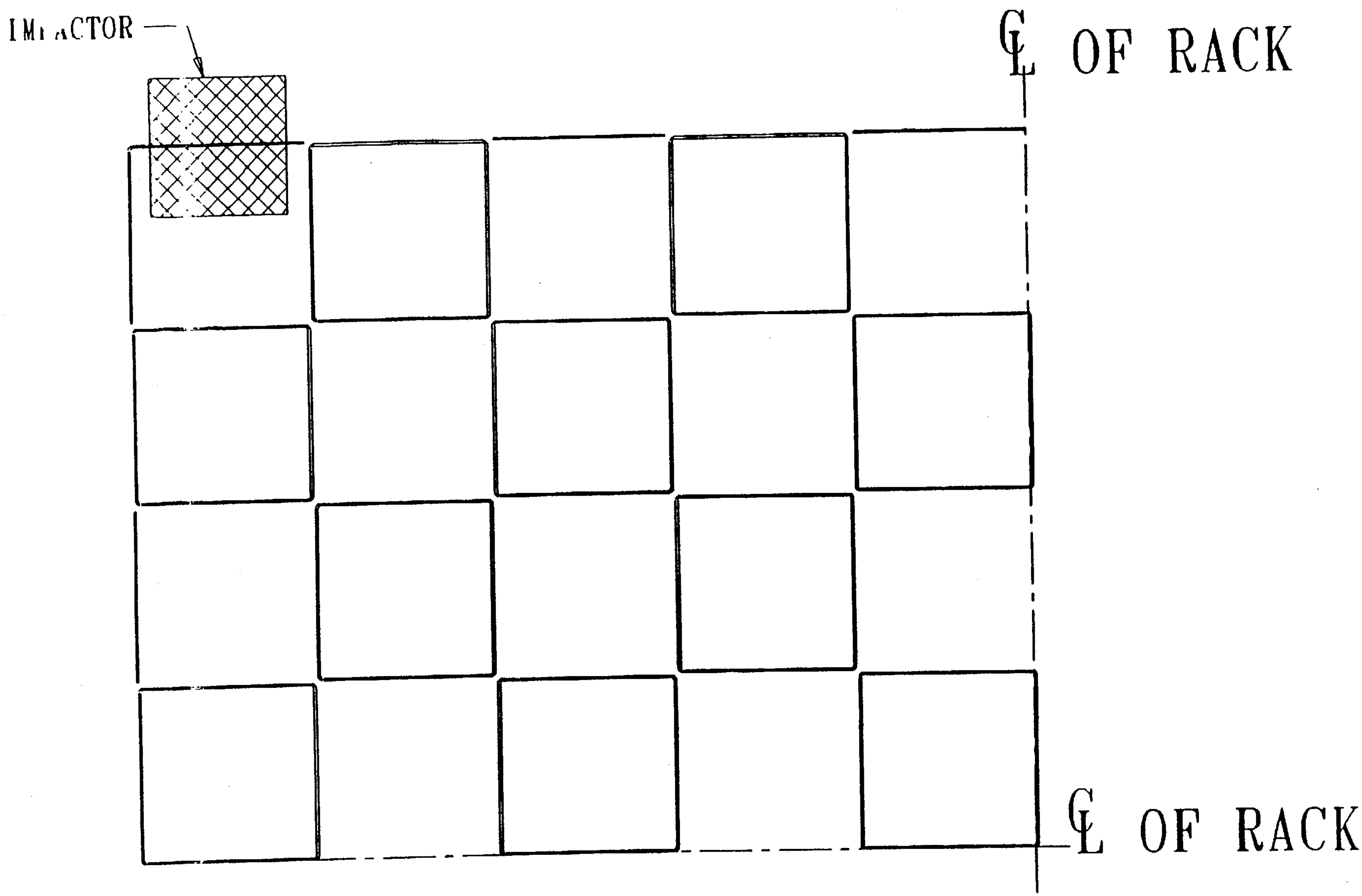


FIGURE 7.2.2; PLAN VIEW OF IMPACTIOR AND IMPACT ZONE
(SHALLOW DROP EVENT)

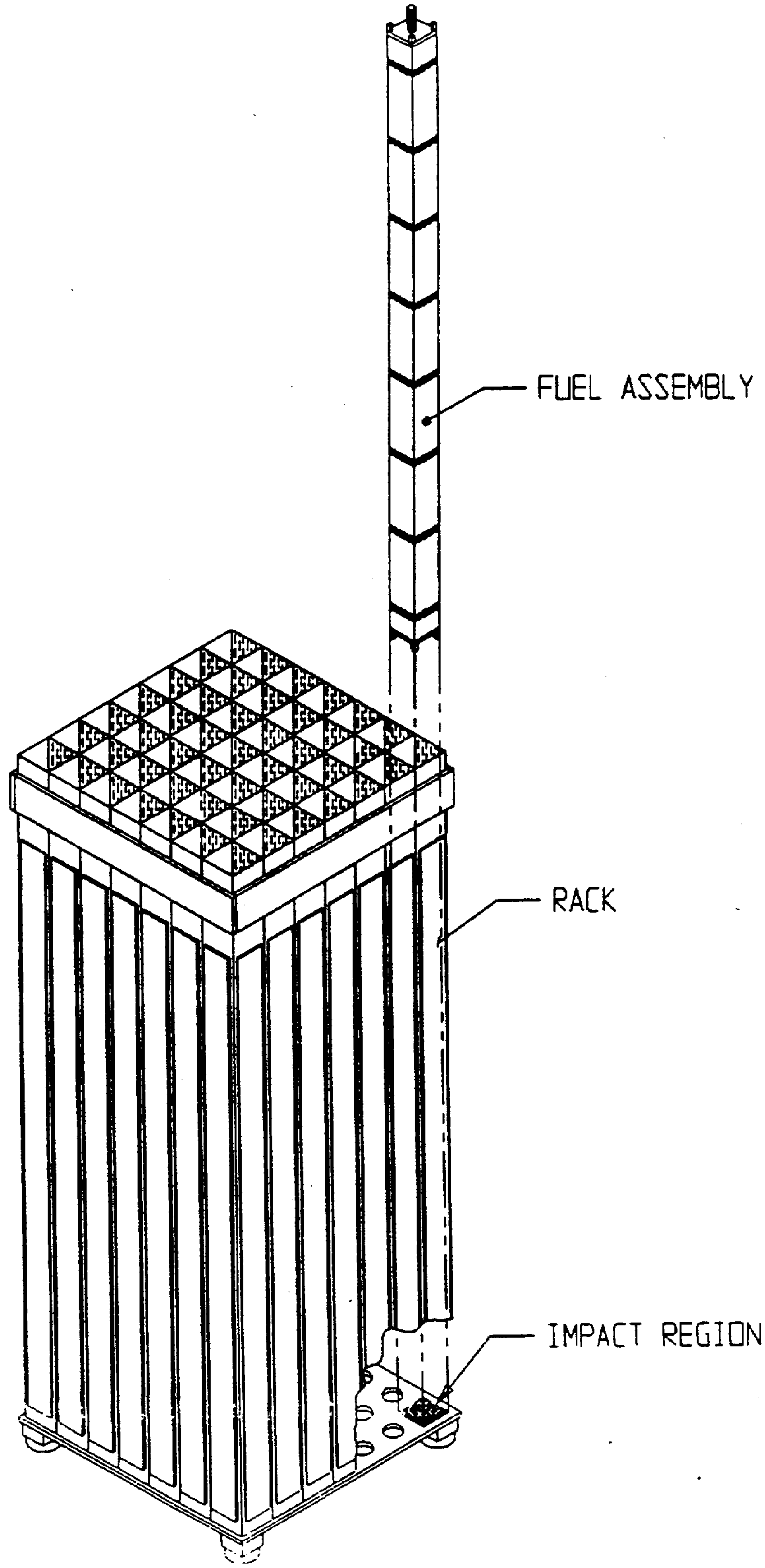


FIGURE 7.2.3; DEEP DROP ON A SUPPORT LEG LOCATION

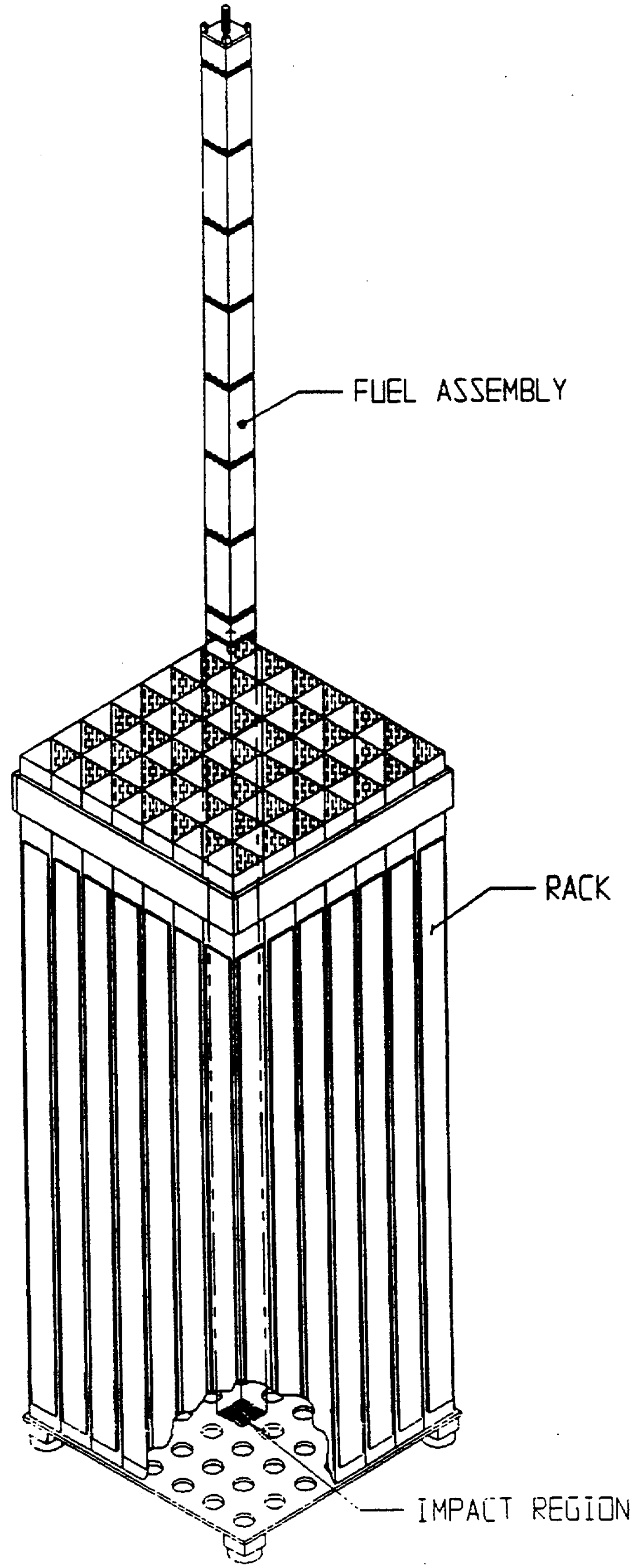


FIGURE 7.2.4; DEEP DROP ON A CENTER CELL LOCATION

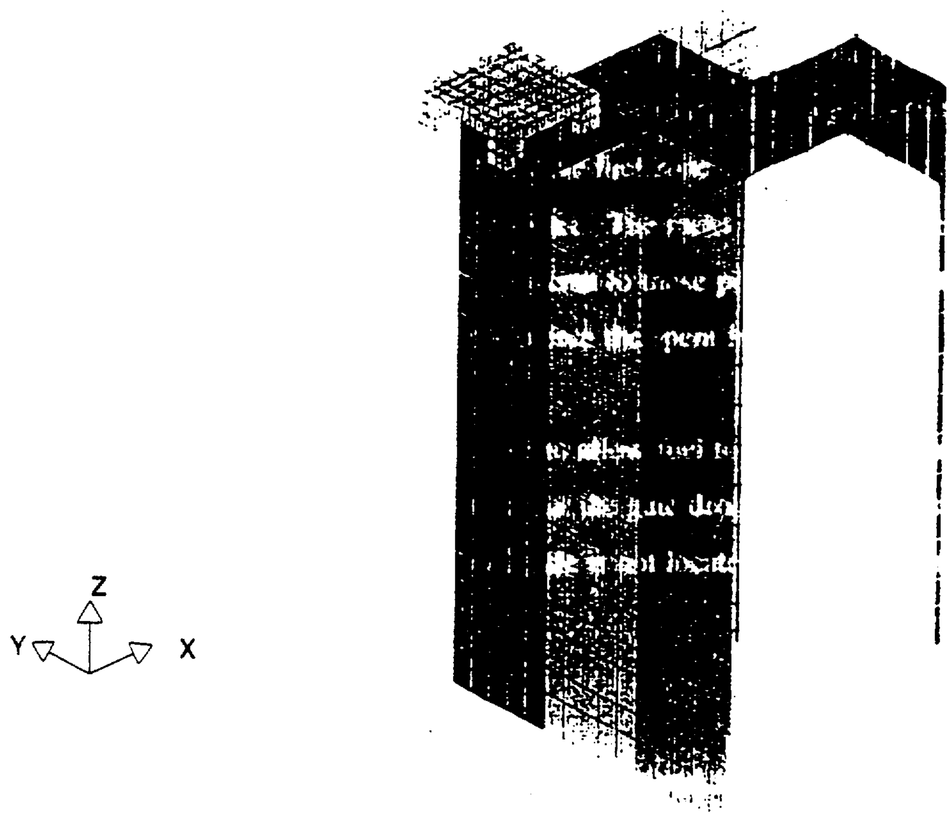


Fig 7.4.1 "Shallow" Drop: Finite-Element Model Detail - Impacted Region

oject 71144

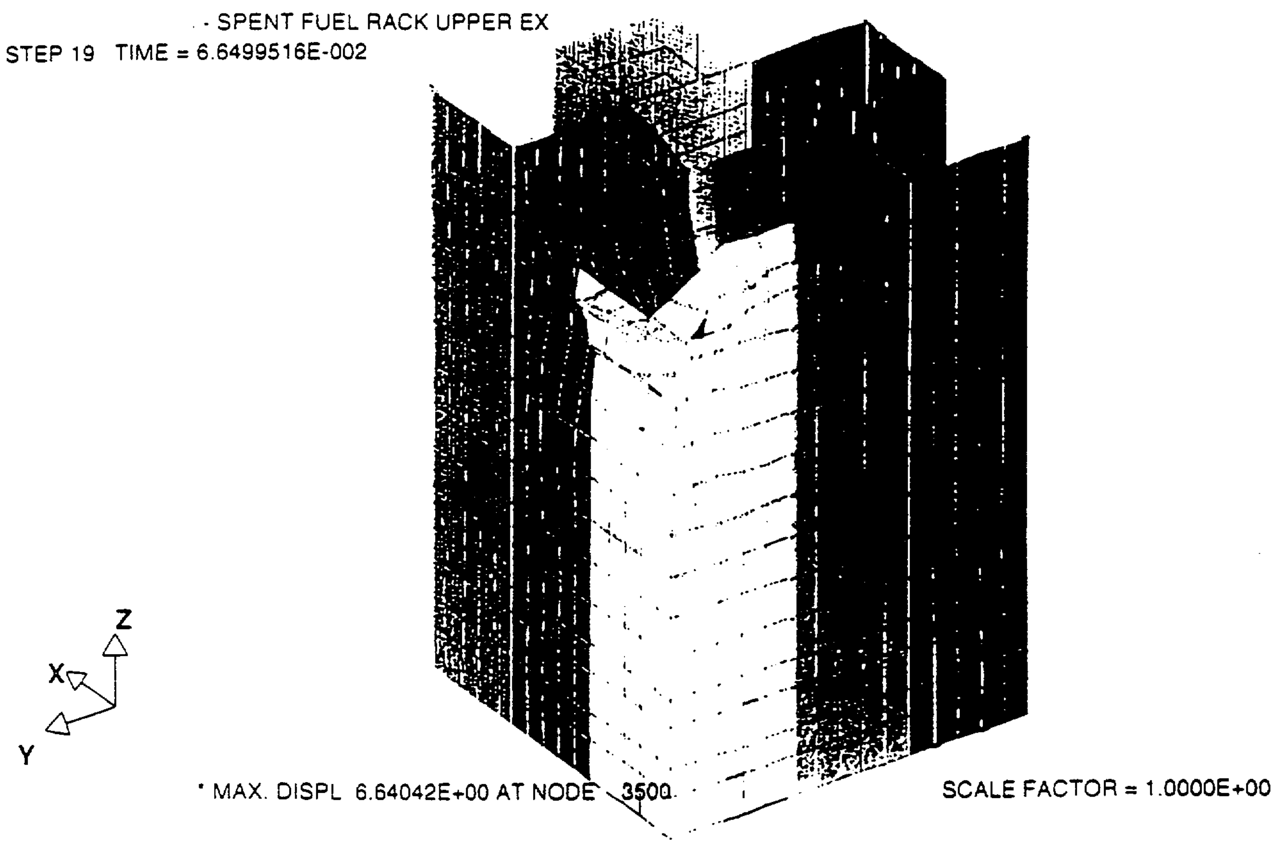


Fig 7 4 2 "Shallow" Drop: Maximum Deformation -

Project 71144

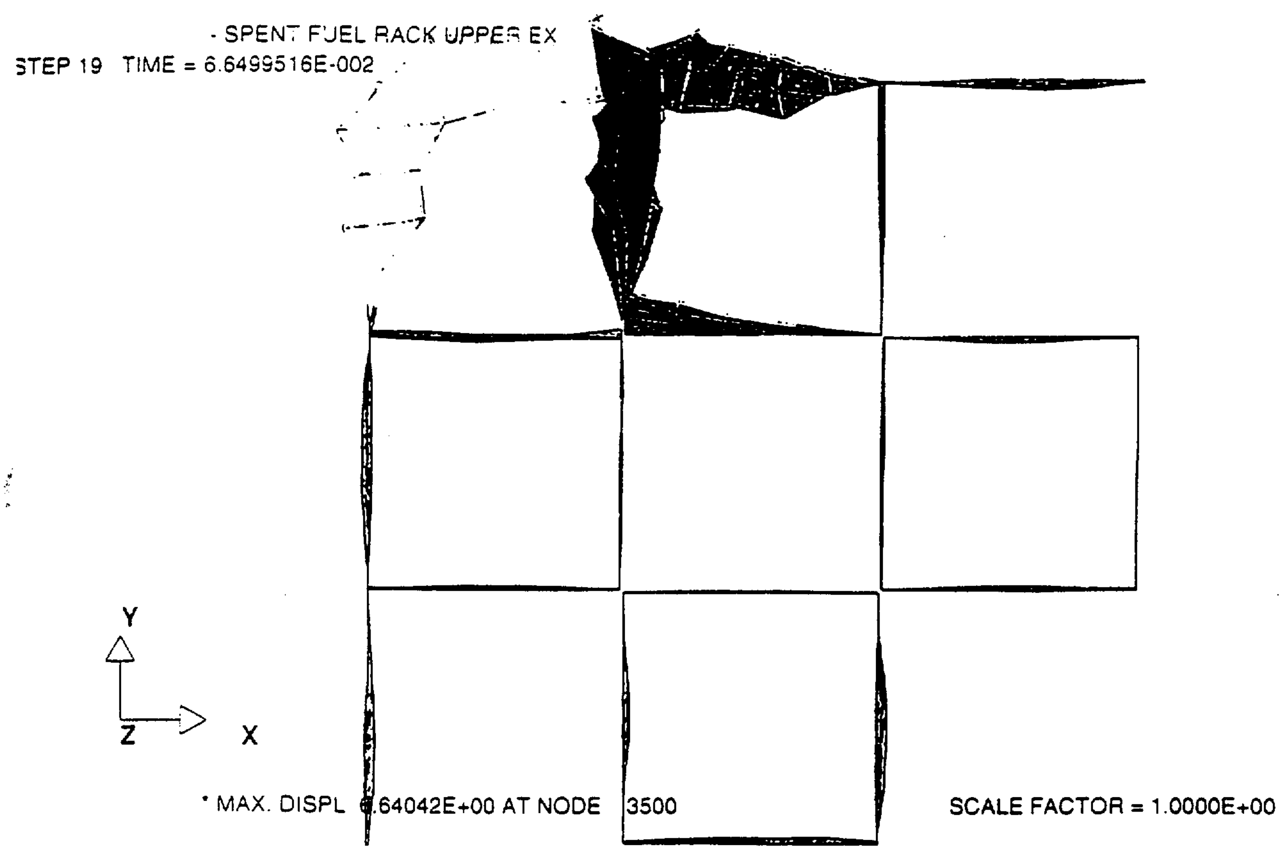
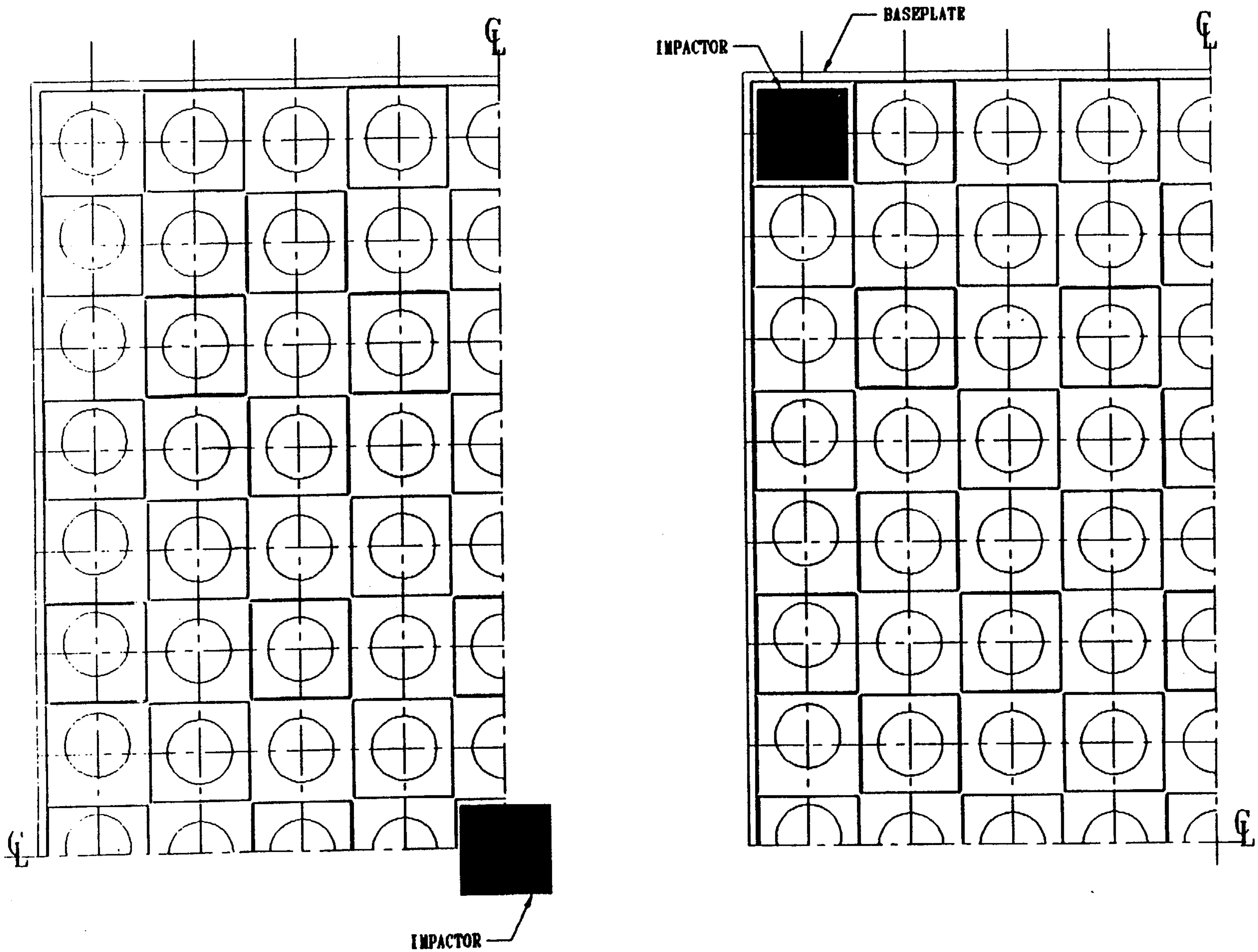


Fig.7.4.3 "Shallow" Drop: Maximum Deformation - Impacted Region (Plan)



(a) *SCENARIO* SC1

(b) *SCENARIO* SC2

FIGURE 7.4.4; PLAN VIEW OF DEEP DROP SCENARIOS

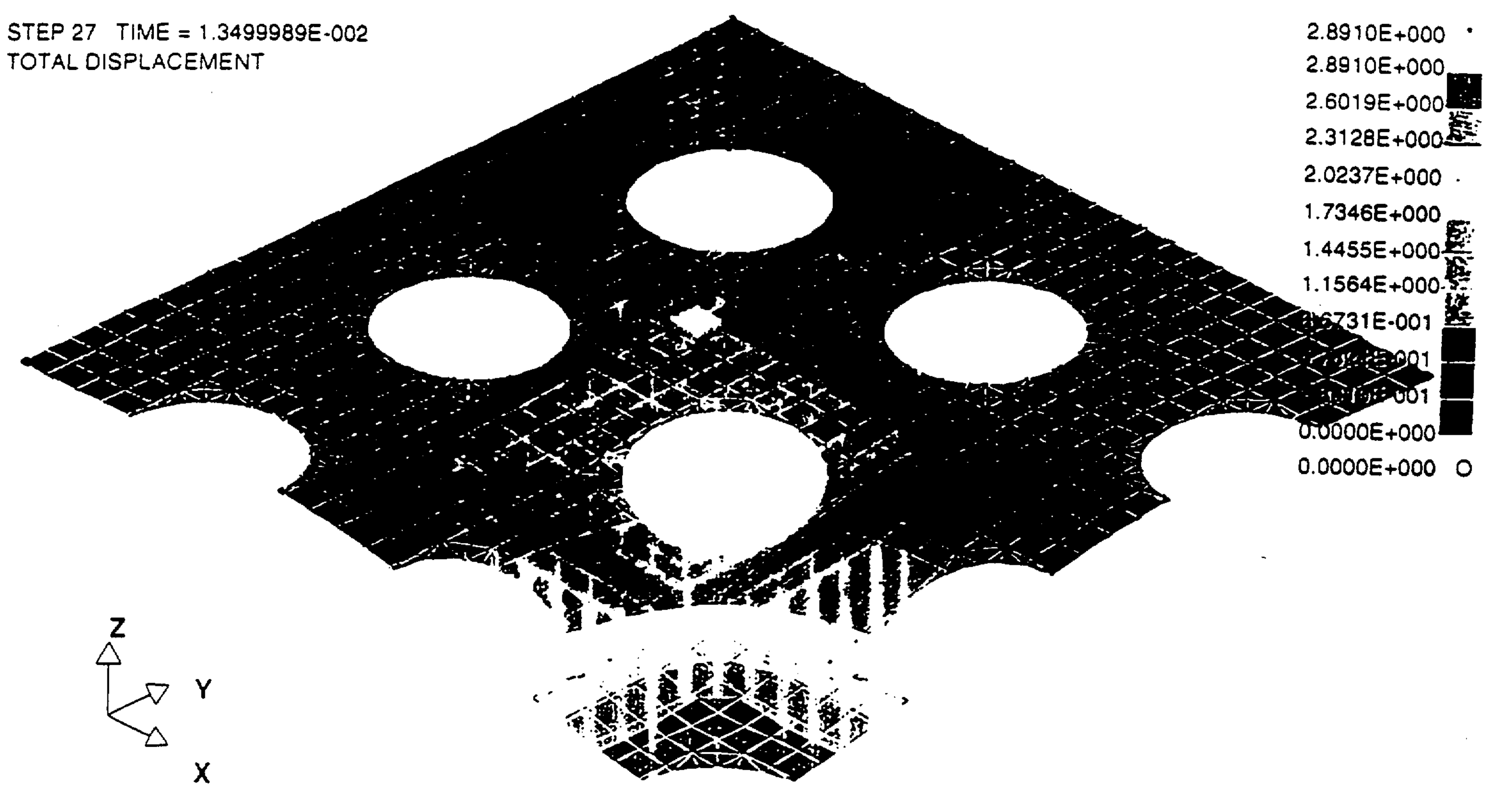


Fig 7 4 5 "Deep" Drop - SC1 Maximum Deformation - Baseplate Region

8.0 FUEL POOL STRUCTURE INTEGRITY CONSIDERATIONS

8.1 Introduction

The MP3 Spent Fuel Pool (SFP) is a safety related, seismic category I, reinforced concrete structure. This section present the analysis to demonstrate structural adequacy of the pool structure, as required by Section IV of the USNRC OT Position Paper [8.1.1].

The pool regions are analyzed using the finite element method. Results for individual load components are combined using factored load combinations mandated by SRP 3.8.4 [8.1.2] based on the "ultimate strength" design method. It is demonstrated that for the critical bounding factored load combinations, structural integrity is maintained when the pools are assumed to be fully loaded with spent fuel racks, as shown in Figure 2.1 with all storage locations occupied by fuel assemblies.

The highly loaded wall sections adjoining the floor slabs are carefully examined. Both moment and shear capabilities are checked for concrete structural integrity. Local punching and bearing integrity of the slab in the vicinity of a rack module support pedestal pad is evaluated. All structural capacity calculations are made using design formulas meeting the requirements of the American Concrete Institute (ACI).

8.2 Description of Pool Structures

The analyzed reinforced concrete structure model is isolated from the remainder of the Fuel Building reinforced concrete structure and includes three pools: the Spent Fuel Pool (SFP), the Cask Pit (CP), and the Transfer Canal (TC). The vertical reinforced concrete walls of the pools are supported at different elevations on a very massive reinforced concrete mat.

The three pools are located in the area delimited by the G and H Fuel Building column lines (parallel to the East direction) and column lines 52.8 and 50.6 (parallel to the North direction) and are separated by reinforced concrete walls of various thicknesses. The walls are supported at different elevations by a massive on-grade reinforced concrete slab, which extends down to the soil elevation -3'-3". Figure 8.2.1 shows these major structural features of the pool.

The thicknesses of the walls surrounding SFP are: 6'-0" at North and East, and 6'-6" at South and on the West side a 6'-6" thick wall (the Canal Wall) separates the SFP from the TC, and in the South-East corner isolation from the CP is realized with two walls, 6'-0" and 5'-5", located respectively at the North and the West side of the CP. The continuity of the SFP West (Canal wall) and CP West walls is interrupted by the existence of the fuel gate openings. The SFP on-grade mat upper elevation is located at elevation 11'-3" and has a thickness of 14'-6".

The thicknesses of the walls surrounding the TC are: 6'-0" at West, North and South. The thickness of the mat is 12'-6" and its upper elevation is located at 9'-3". A sump is located on the south side of the TC and consequently the mat lowers to the elevation of 9'-3".

The CP mat upper elevation is located at 25'-9", but its Pit floor elevation is only 4'-9". The walls of the CP are 5'-0" and 7'-0 thick along the East and South side, respectively.

8.3 Definition of Loads

Pool structural loading involves the following discrete components:

8.3.1 Static Loading (Dead Loads and Live Loads)

1) Dead weight of the modeled concrete structure is calculated considering a density of 150 lb/ft^3

2) Dead weight of the Fuel Building reinforced concrete upper structure;

3) Live Loads such as cranes transmitted by upper building structure;

4) Hydro-static water pressures which vary linearly along the height of the walls.

8.3.2 Seismic Induced Loads

1) The inertial loads generated by seismic events.

2) Hydrodynamic inertia loads due to the contained water mass and sloshing loads (considered in accordance with [8.3.1]) which arise during a seismic event.

3) Hydrodynamic pressures between racks and pool walls caused by rack motion in the pool during a seismic event.

8.3.3 Thermal Loading

Thermal loading is defined by the temperature existing at the faces of the pool concrete walls and slabs. Two thermal loading conditions are evaluated: The normal operating temperature (150°F) and the accident temperature (200°F).

8.4 Analysis Procedures

8.4.1 Finite Element Analysis Model

The finite element model encompasses the entire Spent Fuel Pool and two other reinforced concrete structures located immediately adjacent to the Spent Fuel Pool (the Cask Pit, and the Transfer Canal). The interaction with the rest of the Fuel Building reinforced concrete, which is not included in the finite-element model, is simulated by imposing appropriate boundary conditions. The structural area of interest for the reracking project includes only the SFP which is involved in the fuel storage capacity increase. However, by augmenting the area of interest, by considering in the constructed finite-element model and numerical investigation the additional areas described above, the perturbation induced by the boundary conditions on the stress field distribution for the area of interest is minimized. A finite element 3D view of the structural elements considered in the numerical investigation is shown in Figure 8.4.1.

The preprocessing capabilities of the STARDYNE computer code [8.4.1] are used to develop the 3-D finite-element model. The STARDYNE finite-element model contains 13,209 nodes, 7,252 solid type finite-elements, 3,692 plate type finite-elements and 24 hydro-dynamic masses. Figure 8.4.1 depicts an isometric view of the three-dimensional finite element model without the water and concentrated masses.

The dynamic behavior of the water mass contained in the SFP during a seismic event is modeled according to the guidelines set in TID-7024 [8.3.1]. Neglecting the possibility of water contained in the Transfer Canal is conservative. The loading which would be induced by the hydrostatic pressure would tend to offset the equivalent pressures on the other side of the wall (in the Spent Fuel Pool). The effect of hydrostatic pressure on only one side of this all more than offsets any loading which would be induced from water sloshing.

To simulate the interaction between the modeled region and the rest of the Fuel Building a number of boundary restraints are imposed upon the described finite-element model.

The behavior of the reinforced concrete existing in the structural elements (walls, slab and mat) is considered elastic and isotropic. The elastic characteristics of the concrete are independent of the reinforcement contained in each structural element for the case when the un-cracked cross-section is assumed. This assumption is valid for all load cases with the exception of the thermal loads, where for a more realistic description of the reinforced concrete cross-section including the assumption of cracked concrete is used. To simulate the variation and the degree of cracking patterns, the original elastic modulus of the concrete is modified in accordance with Reference [8.1.3].

8.4.2 Load Application

The structural region isolated from the Fuel Building is numerically investigated using the finite element method. The pool walls and their supporting reinforced concrete mat are represented by a 3-D finite-element model.

The individual loads considered in the analysis are grouped in five categories: dead load (weight of the pool structure, dead weight of the rack modules and stored fuel, dead weight of the reinforced concrete Fuel Building upper structure, crane deadload, and the hydro-static pressure of the contained water), live loads (crane suspended loads), thermal loads (the thermal gradient through the pool walls and slab for normal operating and accident conditions) and the seismic induced forces (structural seismic forces, interaction forces between the rack modules and the pool slab, seismic loads due to self-excitation of the pool structural elements and contained water, and seismic hydro-dynamic interaction forces between the rack modules and the pool walls for both OBE and SSE conditions). The dead and thermal loads are considered static acting loads, while the seismic induced loads are [illegible]e-dependent

The material behavior under a[illegible] [illegible]ditions is described as elastic and isotropic representing the uncracked characteristics [illegible] the structural elements cross-section, with the exception of the thermal load cases where the [illegible]ial elasticity modulus is re[illegible] in order to simulate the variation and the degree of th[illegible] [illegible]k patterns. This approach [illegible] acknowledges the self-relieving nature of the therma[illegible] [illegible]s. The degree of reductio[illegible] the elastic modulus is calculated based on the average ultimate [illegible]city of the particular struc[illegible] [illegible]ment.

The numerical solution (displ[illegible]ents and stresses) for the c[illegible] [illegible]hen the structure was subjected to dead and thermal [illegible]s is a classical static solut[illegible] the time-dependent seismic induced loads the displacemen[illegible] [illegible]d stress field are calculated employing the spectra (shock) method. This method requires a prior modal eigenvector and eigenvalues extraction. Natural frequencies of the 3-D finite-element model are calculated up to the rigid range, considered as greater than 32 Hz. Three independent orthogonal acceleration spectra are applied to the model. The acceleration spectra are considered to act simultaneously in three-directions. The SRSS method is used to sum the similar quantities calculated for each direction.

Results for individual load cases are combined using the factored load combinations discussed below considering two scenarios: first, when the Spent Fuel Pool and the Cask Pit are full of water; second, when only the Spent Fuel Pool is full of water. The combined stress resultants are compared with the ultimate moments and shear capacities of all structural elements pertinent to the Spent Fuel Pool and Cask Pit, which are calculated in accordance with the ACI 349-85 to develop the safety factors.

8.4.3 Load Combinations

The various individual load cases are combined in accordance with the NUREG-0800 Standard Review Plan [8.1.2] requirements with the intent to obtain the most critical stress fields for the investigated reinforced concrete structural elements.

For "Service Load Conditions" the following load combinations are:

- Load Combination No. 1 = 1.4 + 1.7*L

- Load Combination No. 2 = 1.4* D + 1.7*L + 1.9*E

- Load Combination No. 3 = 1.4* D + 1.7*L - 1.9*E

- Load Combination No. 4 = 0.75* (1.4* D + 1.7*L + 1.9*E +1.7*To)

- Load Combination No. 5 = 0.75* (1.4* D + 1.7*L - 1.9*E +1.7*To)

- Load Combination No. 6 = 1.2*D + 1.9*E

- Load Combination No. 7 = 1.2*D - 1.9*E

For "Factored Load Conditions" the following load combinations are:

- Load Combination No. 8 = D + L + To + E'

- Load Combination No. 9 = D + L + To - E'

- Load Combination No. 10 = D + L + Ta + 1.25*E

- Load Combination No. 11 = D + L + Ta - 1.25*E

- Load Combination No. 12 = D + L + Ta + E'

- Load Combination No. 13 = D + L + Ta - E'

where:

- D = dead loads;
- L = live loads;
- To = thermal load during normal operation;
- Ta = thermal load under accident condition;
- E = OBE earthquake induced loads;
- E' = SSE earthquake induced loads.

8.5 Results of Analyses

The STARDYNE postprocessing capability is employed to form the appropriate load combinations and to establish the limiting bending moments and shear forces in various sections of the pool structure. A total of 13 load combinations are computed. Section limit strength formulas for b... ading are computed using appropriate concrete and reinforcement strengths. For MP3, the concrete and reinforcement allowable strengths are:

concrete f_c'	= 5,000 psi
reinforcement f_y	= 60,000 psi

Table 8.5.1 and 8.5.2 shows results from potentially limiting load combinations for the bending strength and shear of the slab and walls, respectively. They demonstrate tl... uctural capacity is not exceeded.

In the tables, a limiting safety margin is defined for each section; the allowable bending moment and shear force defined by ACI divided by the calculated bending moment or shear force (from the finite element analyses). The major regions of the pool structure consist of ten concrete walls delimiting the SFP and Cask Pit. Each area is searched independently for the maximum bending moments in different bending directions and for the maximum shear forces. Safety margins are determined from the calculated maximum bending moments and shear forces based on the local strengths. The procedure is repeated for all the potential limiting load combinations

8.6 Pool Liner

The pool liner is subject to in-plate strains due to movement of the rack support feet during the seismic event. Analyses are performed to establish that the liner will not tear or rupture under limiting loading conditions in the pool, and that there is no fatigue problem under the condition of 1 SSE event plus 20 OBE events. These analyses are based on loadings imparted from the most highly loaded pedestal in the pool assumed to be placed in the most unfavorable position.

8.7 Bearing Pad Analysis

To protect the pool slab from high localized dynamic loadings, bearing pads are placed between the pedestal base and the slab. Fuel rack pedestals impact on these bearing pads during a seismic event and pedestal loading is transferred to the liner. Bearing pad dimensions are set to ensure that the average pressure on the slab surface due to a static load plus a dynamic impact load does not exceed the American Concrete Institute, ACI-349 [8.1.3] limit on bearing pressures. Section 10 of the code gives the design bearing strength as

$$f_b = \Phi (.85\, f_c') \varepsilon$$

where $\Phi = .7$ and f_c' is the specified concrete strength for the spent fuel pool. $\varepsilon = 1$ except when the supporting surface is wider on all sides than the loaded area. In that case, $\varepsilon = (A_2/A_1)^{.5}$, but not more than 2. A_1 is the actual loaded area, and A_2 is an area greater than A_1 and is defined in [8.1.3]. Using a value of $\varepsilon > 1$ includes credit for the confining effect of the surrounding concrete. It is noted that this criteria is in conformance with the ultimate strength primary design methodology of the American Concrete Institute in use since 1971. For MP3 the compressive strength is, $f_c' = 5{,}000$ psi, and the allowable static bearing pressure is $f_b = 2{,}975$ psi assuming no concrete confinement.

The bearing pad selected is 1" thick, austenitic stainless steel plate stock. Most rack pedestals are located away from leak chases. However, in the most limiting configuration, the bearing pad is centered over a leak chase.

An ANSYS finite element simulation of the model is presented in Figure 8.7.1. The model permits the bearing pad to deform and lose contact with the liner, if the conditions of elastostatics so dictate. The slab is modeled as an elastic foundation which supports the liner. A vertical force of 221,000 lbs is applied to the model. This load is chosen to bound the factored results of the rack time-history simulations.

The average pressure at the pad to liner interface is computed and compared against the stress limit. Calculations show that the average pressure at the slab/liner interface is 2,564 psi which is below the ACI allowable of 2,975 psi, providing a factor of safety of 1.16.

The stress distribution in the bearing pad is also evaluated. The maximum bending stress in the bearing pad under the peak vertical load is 21,747 psi. With a material yield strength of 25,000 psi at 200°F, the factor of safety is 1.15.

Therefore, the bearing pad design devised for MP3 is deemed appropriate for the prescribed loadings.

8.8 Conclusions

Regions affected by loading the fuel pool completely with high density racks are examined for structural integrity. It is determined that adequate safety margins exist assuming that all racks are fully loaded with a bounding fuel weight and that the factored load combinations are checked against the appropriate structural design strengths. It is also shown that local loading on the liner does not compromise liner integrity under a postulated fatigue condition and that concrete bearing strength limits are not exceeded.

8.9 References

[8.1.1] OT Position for Review and Acceptance of Spent Fuel Handling Applications, by B.K. Grimes, USNRC, Washington, D.C., April 14, 1978.

[8.1.2] NUREG-0800, SRP-3.8.4, Rev. 1., July 1981.

[8.1.3] ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, Detroit Michigan.

[8.3.1] "Nuclear Reactors and Earthquakes, U.S. Department of Commerce, National Bureau of Standards, National Technical Information Service, Springfield, Virginia (TID 7024).

[8.4.1] STARDYNE User's Manual, Research Engineers, Inc., Rev. 4.4, July 1996.

Table 8.5.1

BENDING STRENGTH EVALUATION

Location	Limiting Safety Margin	Critical Load Combinations (see Section 8.4.3)
Canal Wall	18.01	12
Cask Pit West Wall	13.75	13
Pool East Wall	20.06	13
Pool North Wall	22.16	11
Cask Pit North Wall	19.56	12
Pool South Wall	17.77	12

Table 8.5.2 SHEAR STRENGTH EVALUATION		
Location	Limiting Safety Margin	Critical Load Combinations (see Section 8.4.3)
Canal Wall	4.73	12
Cask Pit West Wall	1.97	12
Pool East Wall	3.39	12
Pool North Wall	2.68	12
Cask Pit North Wall	2.47	13
Pool South Wall	3.6	13

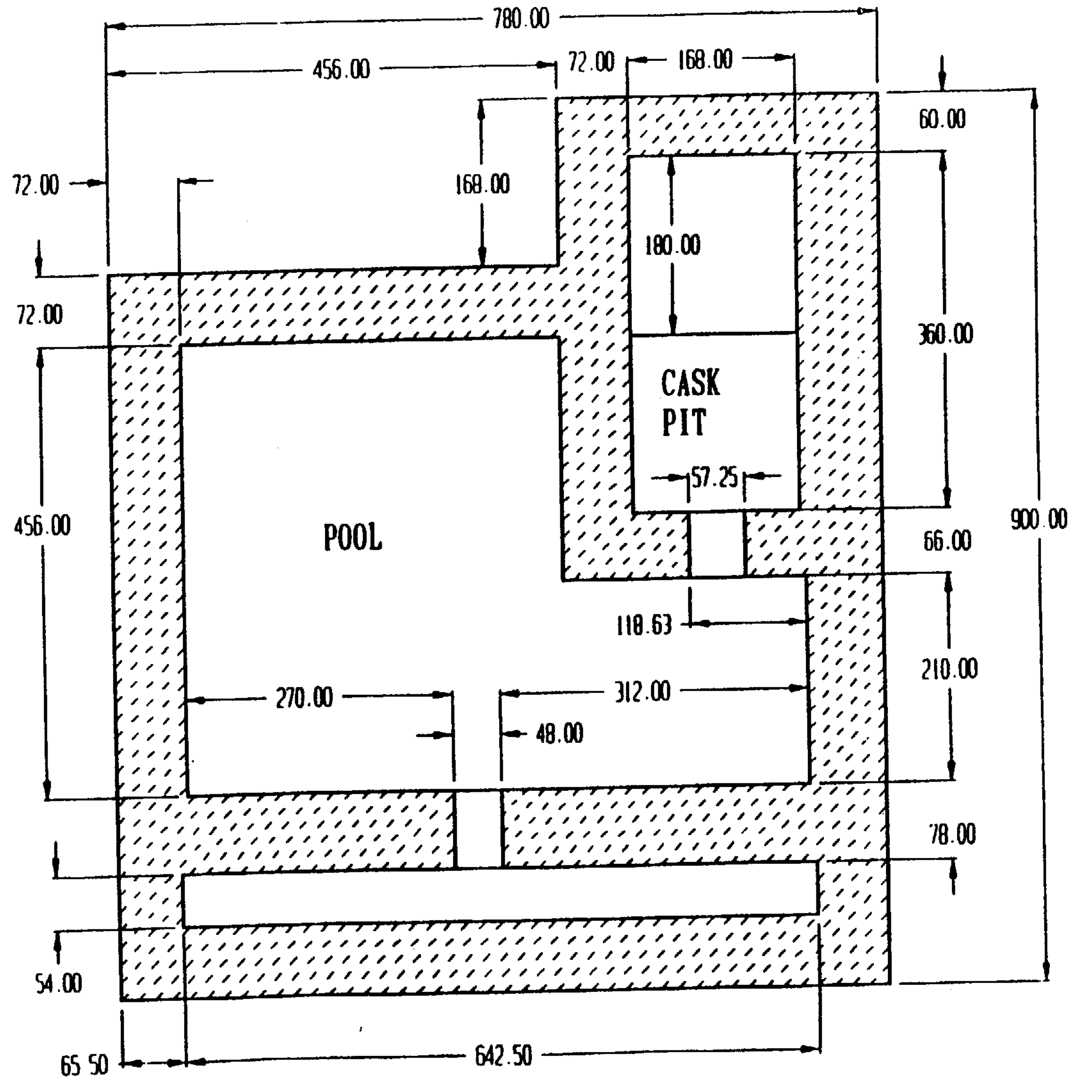


FIGURE 8.2.1: MAJOR STRUCTURAL DIMENSIONS OF MP-3 SPENT FUEL POOL

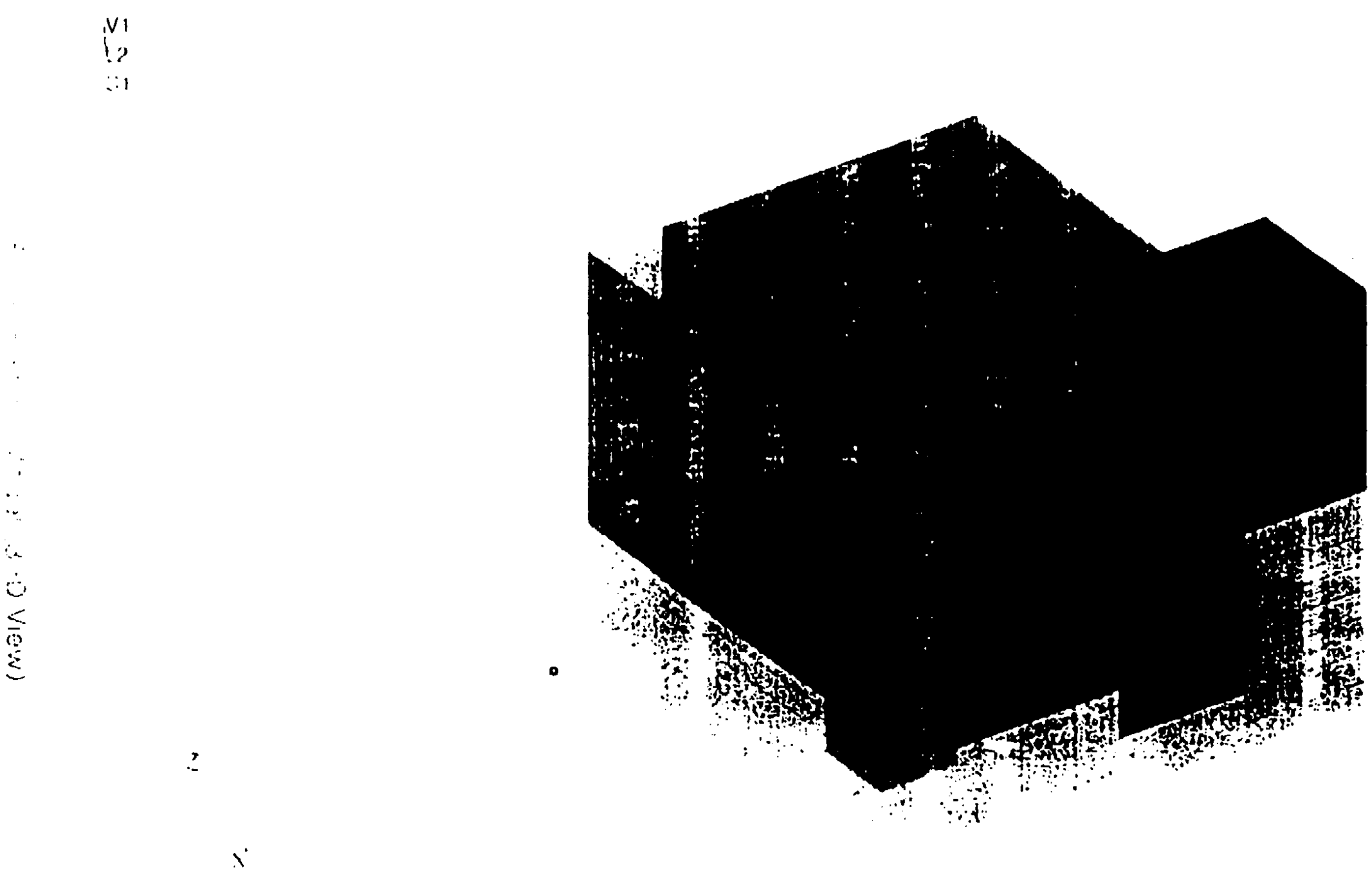


FIGURE 8.4.1

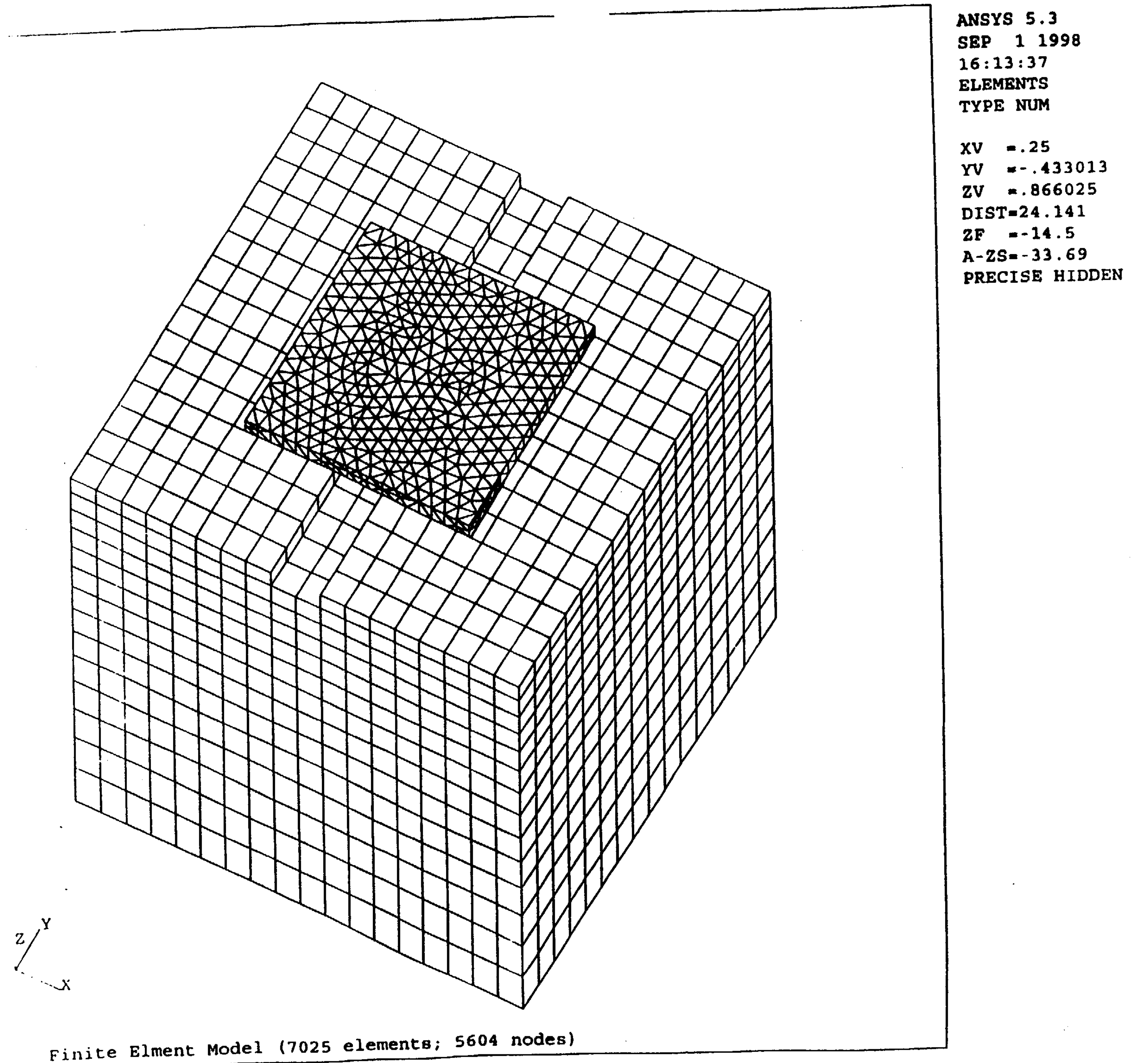

ANSYS 5.3
SEP 1 1998
16:13:37
ELEMENTS
TYPE NUM
XV =.25
YV =-.433013
ZV =.866025
DIST=24.141
ZF =-14.5
A-ZS=-33.69
PRECISE HIDDEN
Z
Y
X
Finite Elment Model (7025 elements; 5604 nodes)

Figure 8.7.1

9.0 BORAL SURVEILLANCE PROGRAM

9.1 Purpose

Boral™, the neutron absorbing material incorporated in the spent fuel storage rack design to assist in controlling system reactivity, consists of finely divided particles of boron carbide (B_4C) uniformly distributed in type 1100 aluminum powder, clad in type 1100 aluminum and pressed and sintered in a hot-rolling process. Tests simulating the radiation, thermal and chemical environment of the spent fuel pool have demonstrated the stability and chemical inertness of Boral (References [9.1.1]-[9.1.3]). The accumulated dose to the Boral over the expected rack lifetime is estimated to be about 3×10^{10} to 1×10^{11} rads depending upon how the racks are used and the number of full-core off-loads that may be necessary.

Based upon the accelerated test programs, Boral is considered a satisfactory material for reactivity control in spent fuel storage racks and is fully expected to fulfill its design function over the lifetime of the racks. Nevertheless, it is prudent to establish a surveillance program to monitor the integrity and performance of Boral on a continuing basis and to assure that slow, long-term synergistic effects, if any, do not become significant. Furthermore, the April 14, 1978 USNRC letter to all power reactor licensees (Reference [9.1.4]), specifies that

> "Methods for verification of long-term material stability and mechanical integrity of special poison materials utilized for neutron absorption should include actual tests."

The purpose of the surveillance program is to characterize certain properties of the Boral with the objective of providing data necessary to assess the capability of the Boral panels in the racks to continue to perform their intended function. The surveillance program is also capable of detecting the onset of any significant degradation with ample time to take such corrective action as may be necessary.

In response to the need for a comprehensive Boral surveillance program to assure that the subcriticality requirements of the stored fuel array are safely maintained, a surveillance program has been developed incorporating certain basic tests and acceptance criteria. The Boral surveillance program depends primarily on representative coupon samples to monitor performance of the absorber material without disrupting the integrity of the storage system. The principal parameters to be measured are the thickness (to monitor for swelling) and boron content.

9.2 COUPON SURVEILLANCE PROGRAM

9.2.1 Coupon Description

The coupon measurement program includes coupons suspended on a mounting (called a "tree"), placed in a designated cell, and surrounded by spent fuel. Coupons will be removed from the array on a prescribed schedule and certain physical and chemical properties measured from which the stability and integrity of the Boral in the storage cells may be inferred.

Each surveillance coupon will be approximately 4 inches wide and 8 inches long. The coupon surveillance program will use a total of 8 test coupons. In mounting the coupons on the tree, the coupons will be positioned axially within the central 8 feet of the fuel zone where the gamma flux is expected to be reasonably uniform.

Each coupon will be carefully pre-characterized prior to insertion in the pool to provide reference initial values for comparison with measurements made after irradiation. The surveillance coupons will be pre-characterized for weight, length, width and thickness. In addition, two coupons will be preserved as archive samples for comparison with subsequent test coupon measurements. Wet chemical analyses of samples from the same lot of Boral will be available from the vendor for comparison.

9.2.2 Surveillance Coupon Testing Schedule

The coupon tree is surrounded by freshly discharged fuel assemblies at each of the first five refuelings following installation of the racks to assure that the coupons will have experienced a slightly higher radiation dose than the Boral in the racks. Beginning with the fifth load of spent fuel, the fuel assemblies will remain in place for the remaining lifetime of the racks. The scheduled coupon management schedule is shown in Table 9.1.

At the time of the first fuel off-load following installation of the coupon tree, the (8) storage cells surrounding the tree shall be loaded with freshly-discharged fuel assemblies that had been among the higher specific power assemblies in the core. Shortly before the second reload, the coupon tree is removed and a coupon removed for evaluation. The coupon tree is then re-installed and, at reload, again surrounded by freshly discharged fuel assemblies. This procedure is continued for the third, fourth, and fifth off-loading of spent fuel (except that a coupon is not pulled at the fourth refueling). From the fifth cycle on, the fuel assemblies in the (8) surrounding cells remain in place.

Evaluation of the coupons removed will provide information of the effects of the radiation, thermal and chemical environment of the pool and by inference, comparable information on the Boral panels in the racks. Over the duration of the coupon testing program, the coupons will have accumulated more radiation dose than the expected lifetime dose for normal storage cells.

Coupons which have not been destructively analyzed by wet-chemical processes, may optionally be returned to the storage pool and re-mounted on the tree. They will then be available for subsequent investigation of defects, should any be found.

9.2.3 Measurement Program

The coupon measurement program is intended to monitor changes in physical properties of the Boral absorber material by performing the following measurements on the pre-planned schedule:

- Visual Observation and Photography.

- Neutron Attenuation,
- Dimensional Measurements (length, width and thickness),
- Weight and Specific Gravity, and
- Wet-chemical analysis (Optional).

The most significant measurements are thickness (to monitor for swelling) and neutron attenuation† (to confirm the concentration of Boron-10 in the absorber material). In the event loss of boron is observed or suspected, the data may be augmented by wet-chemical analysis (a destructive gravimetric technique for total boron only).

9.2.4 Surveillance Coupon Acceptance Criteria

Of the measurements to be performed on the Boral surveillance coupons, the most important are (1) the neutron attenuation measurements (to verify the continued presence of the boron) and (2) the thickness measurement (as a monitor of potential swelling). Acceptance criteria for these measurements are as follows:

- A decrease of no more than 5% in Boron-10 content, as determined by neutron attenuation, is acceptable. (This is tantamount to a requirement for no loss in boron within the accuracy of the measurement.)

- An increase in thickness at any point should not exceed 10% of the initial thickness at that point.

Changes in excess of either of these two criteria requires investigation and engineering evaluation which may include early retrieval and measurement of one or more of the remaining coupons to

† Neutron attenuation measurements are a precise instrumental method of chemical analysis for Boron -10 content using a non-destructive technique in which the per centage of thermal neutrons trans mitted through the panel is measured and compared with pre- deter mined calibration data. Boron-10 is the nuclide of principal interest since it is the isotope responsible for neutron absorption in the Boral panel.

provide corroborative evidence that the indicated change(s) is real. If the deviation is determined to be real, an engineering evaluation shall be performed to identify further testing or any corrective action that may be necessary.

The remaining measurement parameters serve a supporting role and should be examined for early indications of the potential onset of Boral degradation that would suggest a need for further attention and possibly a change in measurement schedule. These include (1) visual or photographic evidence of unusual surface pitting, corrosion or edge deterioration, or (2) unaccountable weight loss in excess of the measurement accuracy.

9.3 In-Service Inspection (Blackness Tests)

In-service inspection involves directly testing the Boral panels in the storage racks by neutron logging[+] (sometimes called "Blackness Testing"). This technique is able to detect areas of significant boron loss or the existence of gaps in the Boral, but cannot determine other physical properties such as those measured in the coupon program.

In the event that the surveillance coupon program shows a confirmed indication of degradation, blackness testing may be one of the techniques employed to investigate the extent of degradation, if any, in the racks.

[+] Neutron logging, is a derivative of well-logging methods successfully used in the oil industry for many years.

9.4 References

[9.1.1] "Spent Fuel Storage Module Corrosion Report", Brooks & Perkins Report 554, June 1, 1977

[9.1.2] "Suitability of Brooks & Perkins Spent Fuel Storage Module for Use in PWR Storage Pools", Brooks & Perkins Report 578, July 7, 1978

[9.1.3] "Boral Neutron Absorbing/Shielding Material - Product Performance Report", Brooks & Perkins Report 624, July 2O, 1982

[9.1.4] USNRC Letter to All Power Reactor Licensees, transmitting the "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978

Table 9.1

COUPON MEASUREMENT SCHEDULE

[redacted]

- * Remove coupons for ev... the next refueling. The first coupon is schedule... 8.

- ** Place freshly disch... ning of the 1st, 2nd, 3rd, 4th, and 5th refueling cycles after completion of reracking.

10.0 INSTALLATION

10.1 Introduction

All construction work at Millstone 3 will be performed in compliance with NUREG-0612 (refer to Section 3.0), applicable Quality Assurance procedures, and site-specific project procedures.

Crane and fuel bridge operators are to be adequately trained in the operation of load handling machines per the requirements of ANSI/ASME B30.2, latest revision, and the Northeast Utilities training program. Consistent with past practices, videotaped aided training will be given to the installation team, all of whom will be required to successfully complete a written examination prior to the commencement of work.

The lifting device designed for handling and installation of the new racks at Millstone 3 is remotely engageable. The lifting device complies with the provisions of ANSI 14.6-1978 and NUREG-0612, including compliance with the primary stress criteria, load testing at a multiplier of maximum working load, and nondestructive examination of critical welds.

An intensive surveillance and inspection program shall be maintained throughout the installation phase of the rerack project. A complete set of operating procedures which cover the entire gamut of operations pertaining to the rack installation will be used. Similar procedures have been utilized and successfully implemented by Holtec International on previous rack installation projects. These procedures assure ALARA practices are followed and provide detailed requirements to assure equipment, personnel, and plant safety. The following is a list of procedures which will be used to implement the construction phase of the rerack project.

A. Installation/Handling/Removal Procedure:

This procedure provides direction for the handling/installation of the new high density modules. This procedure delineates the steps necessary for receiving a new high density rack on site, the proper method for unloading and uprighting the rack, staging the rack prior to installation, and installation of the rack. This procedure also provides for the installation of new rack bearing pads, adjustment of the new rack pedestals and performance of the as-built field survey. Any pool modifications that may be necessary such as protrusion truncation are also described in the procedure.

B. Receipt Inspection Procedure:

This procedure delineates the steps necessary to perform a thorough receipt inspection of a new rack module after its arrival on site. The receipt inspection includes dimensional measurements, cleanliness inspection, visual weld examination, and verticality measurements.

C. Cleaning Procedure:

This procedure provides for the cleaning of a new rack module, if it is required, in order to meet the requirements of ANSI 45.2.1, Level C. Permissible cleaning agents, methods and limitations on materials to be employed are provided.

D. Pre-Installation Drag Test Procedure:

This procedure stipulates the requirements for performing a functional test on a new rack module prior to installation into the spent fuel pool. The procedure provides direction for inserting and withdrawing a "dummy" fuel assembly into designated cell locations, and establishes an acceptance criteria in terms of maximum kinetic drag force.

E. Post-Installation Drag Test Procedure:

This procedure stipulates the requirements for performing a functional test on a new rack module following installation into the spent fuel pool or cask pit. The procedure will provide direction for inserting and withdrawing a " dummy" fuel assembly into designated cell locations. and establishes an acceptance criteria in terms of maximum kinetic drag force.

F. Underwater Diving Procedure:

Underwater diving operations may be required to support the new rack installation. This procedure describes the method for introducing a diver into the spent fuel pool or cask pit, provides for radiological monitoring during the operation, and defines the egress of the diver from the fuel pool following work completion. Furthermore, this procedure requires strict compliance with OSHA Standard 29CFR-1910, Subpart T. and establishes contingencies in the event of an emergency.

G. ALARA Procedure:

Consistent with the site's ALARA Program, this procedure provides details to minimize the total man-rem received during the rerack project, by accounting for time, distance, and shielding. Additionally, a pre-job checklist is established in order to mitigate the potential for an overexposure.

H. Liner Inspection Procedure:

In the event that a visual inspection of any submerged portion of the spent fuel pool liner is deemed necessary, this procedure describes the method to perform such an inspection using an underwater camera and describes the requirements for documenting any observations.

I. Leak Detection Procedure:

This procedure describes the method to test the spent fuel pool liner for potential leakage using a vacuum box. This procedure may be applied to any suspect area of the pool liner.

J. Underwater Welding Procedure:

In the event of a positive leak test result, an underwater welding procedure will be implemented which will provide for the placement of a stainless steel repair patch over the area in question. The procedure contains appropriate qualification records documenting relevant variables, parameters, and limiting conditions. The weld procedure is qualified in accordance with AWS D3.6-93, Specification for Underwater Welding or may be qualified to an alternate code accepted by Northeast Utilities.

K. Job Site Storage Procedure:

This procedure establishes the requirements for safely storing a new rack module on-site, in the event that long term job-site storage is necessary. This procedure provides environmental restrictions, temperature limits, and packaging requirements.

10.2 Rack Arrangement

The existing Millstone Unit 3 rack arrangement consists of 21 racks, representing 756 cell locations. The new proposed rack arrangement consists of 15 free-standing Holtec racks providing a total of 1,104 storage locations in the fuel pool. Of these 1,104 cell locations, five racks consisting of 350 cells are designated as Region 1 storage, and the remaining 10 racks containing 754 cells are designated as Region 2 storage.

A schematic depicting the spent fuel pool in the new maximum density configuration can be seen in Figure 2.1.

10.3 Pool Survey and Inspection

A pool inspection shall be performed to determine if any items attached to the liner wall or floor will interfere with the placement of the new racks or prevent usage of any cell locations subsequent to installation.

In the event that protrusions are found which would pose any interference to the installation process, it is anticipated that underwater diving operations and mechanical cutting methods would be employed to remove the protrusions.

10.4 Pool Cooling and Purification

10.4.1 Pool Cooling

The spent fuel pool cooling system shall be operated in order to maintain the pool water temperature at an acceptable level. It is anticipated that specific activities, such as bearing pad elevation measurements, may require the temporary shutdown of the spent fuel pool cooling system. At no time, however, will pool cooling be terminated in a manner or for a duration which would create a violation of the Millstone 3 Technical Specification.

Existing procedures are in place to control actions regarding the shutdown of the spent fuel pool cooling system, and to ensure that the pool bulk temperature will always remain within required limits.

10.4.2 Pool Purification

The existing spent fuel pool filtration system shall be operational in order to maintain pool clarity. Additionally, an underwater vacuum system shall be used as necessary to supplement fuel pool purification. A vacuum system may be employed to remove extraneous debris, reduce general contamination levels prior to diving operations, and to assist in the restoration of pool clarity following any hydrolasing operations.

10.5 Installation of New Racks

The new high density racks, supplied by Holtec International, shall be delivered in the horizontal position. A new rack module shall be removed from the shipping trailer using a suitably rated crane, while maintaining the horizontal configuration, and placed upon the upender and secured. Using two independent overhead hooks, or a single overhead hook and a spreader beam, the module shall be uprighted into vertical position.

The new rack lifting device shall be installed into the rack and each lift rod successively engaged. Thereafter, the rack shall be transported to a pre-leveled surface where the appropriate quality control receipt inspection shall be performed.

In preparing the spent fuel pool for rack installation, the pool floor shall be inspected and any debris which may inhibit the installation of bearing pads will be removed.

After pool floor preparation, new rack bearing pads shall be positioned in preparation for the module which is to be installed. Elevation measurements will then be performed in order to gage the amount of adjustment required, if any, for the new rack pedestals.

The new rack module shall be lifted with the 10-ton crane and transported along the safe load path. The rack pedestals shall be adjusted in accordance with the bearing pad elevation measurements in order to achieve module levelness after installation.

The rack modules shall be lowered into the spent fuel pool using another 10-ton crane. A hoist with equivalent capacity may be attached to this crane for installation activities in order to eliminate contamination of the main hook during lifting operations in the pools. The rack shall be carefully lowered onto its bearing pads. Movements along the pool floor shall not exceed six inches above the liner or a height to allow for clearance over floor projections.

Elevation readings shall be taken to confirm that the module is level and as-built rack-to-rack and rack-to-wall offsets shall be recorded. The lifting device shall be disengaged and removed from the fuel pool under Health Physics direction. Post-installation free path verification will be performed using an inspection gage in order to ensure that no cell location poses excessive resistance to the insertion or withdrawal of a bundle. This test shall confirm final acceptability of a new rack module.

10.6 Safety, Radiation Protection, and ALARA Methods

10.6.1 Safety

During the construction phase of the rerack project, personnel safety is of paramount importance, outweighing all other concerns. All work shall be carried out in strict compliance with applicable approved procedures.

10.6.2 Radiation Protection

Health Physics shall provide necessary coverage in order to provide radiological protection and monitor dose rates. The Health Physics department shall prepare Radiation Work permits (RWPs) that will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels, and dosimetry requirements.

In addition, no activity within the radiologically controlled area shall be carried out without the knowledge and approval of Health Physics. Health Physics shall also monitor items removed from the pool or provide for the use of alarming dosimetry and supply direction for the proper storage of radioactive material.

10.6.3 ALARA

The key factors in maintaining project dose As Low As Reasonably Achievable (ALARA) are time, distance, and shielding. These factors are addressed by utilizing many mechanisms with respect to project planning and execution.

Time

Each member of the project team will be properly trained and will be provided appropriate education and understanding of critical evolutions. Additionally, daily pre-job briefings will be employed to acquaint each team member with the scope of work to be performed and the proper means of executing such tasks. Such pre-planning devices reduce worker time within the radiologically controlled area and, therefore, project dose.

Distance

Remote tooling such as lift fixtures, pneumatic grippers, a support leveling device and a lift rod disengagement device have been developed to execute numerous activities from the pool surface, where dose rates are relatively low. For those evolutions requiring diving operations, diver movements shall be restricted by an umbilical, which will assist in maintaining a safe distance from irradiated sources. By maximizing the distance between a radioactive sources and project personnel, project dose is reduced.

Shielding

During the course of the rerack project, primary shielding is provided by the water in the spent fuel pool. The amount of water between an individual at the surface (or a diver in the pool) and an irradiated fuel assembly is an essential shield that reduces dose. Additionally, other shielding, may be employed to mitigate dose when work is performed around high dose rate sources.

10.7 Radwaste Material Control

Radioactive waste generated from the rerack effort shall include vacuum filter bags, miscellaneous tooling, underwater appurtenances and protective clothing.

Vacuum filter bags shall be removed from the pool and stored as appropriate in a suitable container in order to maintain low dose rates. Contaminated tooling shall be properly stored per Radiation Protection direction throughout the project. At project completion, an effort will be made to decontaminate tooling to the most practical extent possible.

11.0 RADIOLOGICAL EVALUATION

11.1 Solid Radwaste

No significant increase in the volume of solid radioactive wastes is expected from operating with the expanded storage capacity. The necessity for pool filtration resin replacement is determined primarily by the requirement for water clarity, and the resin is normally changed about once a year. During re-racking operations, a small amount of additional resins may be generated by the pool cleanup system on a one-time basis.

11.2 Gaseous Releases

Gaseous releases from the fuel storage area are combined with other plant exhausts. Normally, the contribution from the fuel storage area is negligible compared to the other releases and no significant increases are expected as a result of the expanded storage capacity.

11.3 Personnel Doses

During normal operations, personnel working in the fuel storage area are exposed to radiation from the spent fuel pool. Radiological conditions are dominated by the most recent batch of discharged spent fuel. The radioactive inventory of the older fuel is insignificant compared to that from the recent offload. Analysis shows that the rerack will not significantly change radiological conditions. Therefore the rack expansion project falls within the existing design basis of Millstone's Spent Fuel Pool.

11.4 Anticipated Dose During Re-racking

All of the operations involved in re-racking will utilize detailed procedures prepared with full consideration of ALARA principles. Similar operations have been performed in a number of facilities in the past, and there is every reason to believe that re-racking can be safely and efficiently accomplished at MP3, with low radiation exposure to personnel.

Total dose for the re-racking operation is estimated to be between 2 and 5 person-rem, as indicated in Table 11.4.1. While individual task efforts and doses may differ from those in Table 11.4.1, the total is believed to be a reasonable estimate for planning purposes. Table 11.4.2 shows previous job exposures that Holtec International has experienced during actual rack installations. Divers will be used where necessary, and the estimated person-rem burden includes a figure for their possible dose.

The existing radiation protection program at MP3 is adequate for the re-racking operations. Where there is a potential for significant airborne activity, continuous air monitors will be in operation. Personnel will wear protective clothing as required and, if necessary, respiratory protective equipment. Activities will be governed by a Radiation Work Permit, and personnel monitoring equipment will be issued to each individual. As a minimum, this will include thermoluminescent dosimeters (TLDs) and self-reading dosimeters. Additional personnel monitoring equipment (i.e., extremity TLDs or multiple TLDs) may be utilized as required.

Work, personnel traffic, and the movement of equipment will be monitored and controlled to minimize contamination and to assure that dose is maintained ALARA.

Table 11.4.1

PRELIMINARY ESTIMATE OF PERSON-REM DOSE DURING RE-RACKING

Step	Number of Personnel	Hours	Estimated Person-Rem Dose†
Clean and vacuum pool	3	25	0.3 to 0.6
Remove underwater appurtenances	4	80	0.4 to 0.8
Installation of new rack modules	5	55	0.7 to 1.3
Move fuel to new racks	2	150	0.8 to 1.5
Total Dose, person-rem			2 to 5

† Assumes minimum does rate of 2-1/2 mrem/hr (expected) to a maximum of 5 mrem/hr except for pool vacuuming operations, which assume 4 to 8 mrem/hr, and diving operations, which assume 20 to 40 mrem/hr.

Table 11.4.2

SPENT FUEL RERACK EXPOSURE

Plant	Job Exposure (Man-Rem)
TMI	5.9
D.C. Cook	2.2
Ft. Calhoun	2.5
Zion	13.0*
Salem Unit 1/ Unit 2	4.5/1.0
Limerick	2.0
Duane Arnold	5.5
Connecticut Yankee	7.5
Sequoyah	2.5

* N.B. Hydrolasing was not permitted to maintain Boron concentration levels in the pool. Existing racks were removed and steam cleaned in the decon pit.

12.0 ENVIRONMENTAL COST-BENEFIT ASSESSMENT

12.1 Introduction

Article V of the USNRC OT Position paper [12.1] requires the submittal of a cost/benefit analysis for the chosen fuel storage capacity enhancement method. This section abstracts the analyses and evaluations made by NU before selecting reracking as the most viable alternative.

12.2 Imperative for Reracking

The specific need to increase the limited existing storage capacity of the MP-3 spent fuel pool is based on the continually increasing inventory in the pool, the prudent requirement to maintain full-core off-load capability, and a lack of viable economic alternatives. In particular:

a. NNECO has no current contractual arrangements with fuel reprocessing facilities, nor is this technology economically viable in the U.S.

b. NUSCO (on behalf of Millstone Unit 3) has executed a disposal contract with the Department of Energy (DOE) pursuant to the Nuclear Waste Policy Act of 1982, but DOE has no plans to provide disposal facilities prior to 2010.

c. 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 will be sent offsite for final disposition under existing legislation, but (as indicated above) the government facility is not expected to be available to begin to receive fuel for at least 12 years.

Reference is made to Tables 1.1 and 1.2 of Section 1 wherein the current and projected fuel discharges in the MP-3 spent fuel pool are tabulated. It is seen that the MP-3 fuel pool will lose the capacity to discharge one full core (193 fuel assemblies) in 2000.

12.3 Appraisal of Alternative Options

NU has determined that wet storage expansion is by far the most viable option for the MP-3 pool in comparison to other alternatives.

The key considerations in evaluating the alternative options were:

- Safety: minimize the number of fuel handling steps
- Economy: minimize total installed and O&M cost
- Security: protection from potential saboteurs, natural phenomena
- Non-intrusiveness: minimize required modification to existing systems
- Maturity: extent of industry experience with the technology
- ALARA: minimize cumulative dose due to handling of fuel.

Wet storage expansion was found by NNECO to be the most attractive option with respect to each of the foregoing criteria. In particular:

a. There are no operational commercial interim storage facilities available for NNECO's needs in the United States, nor are there expected to be any in the foreseeable future.

b. While plans are being formulated by DOE for construction of a spent fuel repository pursuant to the Nuclear Waste Policy Act of 1982, this facility is not expected to be available to accept spent fuel any earlier than 2010. Furthermore, DOE's Acceptance Priority Rankings suggest that Millstone-3's spent fuel would be removed substantially later than 2010.

c. Dry storage could be a technically feasible alternative to wet storage. However, the least expensive type of dry storage has been evaluated to entail a capital expenditure that is approximately 3.5 times as large as that associated with wet storage. Other problems with dry storage include substantial incremental fuel movements, storage located away from the secured boundary of the site, incremental security requirements and operation and maintenance expenses, plant modifications to support the use of dry storage cask systems, and potential repackaging of fuel to meet repository requirements.

To summarize, the only acceptable option for Millstone Unit 3 is to increase its onsite wet fuel storage capacity. The alternatives have either little proven experience, or they are cost prohibitive.

12.4 Cost Estimate

The proposed construction contemplates the reracking of the MP-3 spent fuel pool using free-standing, high density, poisoned spent fuel racks. The engineering and design is completed for full reracking of the MP-3 pool. This rerack will provide sufficient MP-3 pool storage capacity to maintain a full core off-load capability to approximately the end of license.

The total capital cost is estimated to be approximately $10 million as detailed below. Cost estimates do not include cost of capital, overhead, or project contingencies. They are for the purpose of comparison only.

Engineering, design. project management	$2 million
Rack fabrication	$5 million
Rack installation	$3 million

As described in the preceding section, many alternatives were considered prior to proceeding with wet storage expansion, which is not the only technical option available to increase on-site storage capacity. Wet storage expansion does, however, provide a definite cost advantage over other technologies.

12.5 Resource Commitment

The expansion of the MP-3 spent fuel pool capacity is expected to require the following primary resources:

Stainless steel:	250 tons
Boral neutron absorber:	60 tons, of which 50 tons is Boron Carbide powder and 10 tons are aluminum.

The requirements for stainless steel and aluminum represent a small fraction of total world output of these metals (less than 0.001%). Although the fraction of world production of Boron Carbide required for the fabrication is somewhat higher than that of stainless steel or aluminum, it is unlikely that the commitment of Boron Carbide to this project will affect other alternatives. Experience has shown that the production of Boron Carbide is highly variable and depends upon need and can easily be expanded to accommodate worldwide needs.

12.6 Environmental Considerations

This rerack is not expected to increase the maximum bulk pool temperature above the previously licensed value. Therefore, the cooling water demand on the Long Island Sound and the water vapor emission to the environment should remain unchanged.

12.7 References for Section 12

[12.1] OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications, USNRC (April, 1978).

[12.2] Electric Power Research Institute, Report No. NF-3580, May, 1984.

[12.3] "Spent Fuel Storage Options: A Critical Appraisal", Power Generation Technology, Sterling Publishers, pp. 137-140, U.K. (November, 1990).

Docket No. 50-423
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Attachment 6

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specification
Spent Fuel Pool Rerack (TSCR 3-22-98)

Proprietary Version of
Licensing Report for Spent Fuel Rack Installation
at Millstone Nuclear Station Unit 3

NOT PRODUCED

March 1999